DOCUMENT RESUME

ED 192 149 CE 026 780

TITLE	Nuclear Technology Series. Course 9: Reactor Auxiliary Systems.
INSTITUTION	Center for Occupational Research and Development, Inc., Waco, Tex.: Technical Education Research Center, Waco, Tex.
SPONS AGENCY	Office of Education (DHEW), Washington, D.C.
PUB DATE	Jun 80
CONTRACT	300-76-0356
NOTE	102p.: For related documents see CE 026 639-672, CE 026 779-780, CE 026 782-785, and CE 026 855-856.
AVAILABLE FROM	Center for Occupational Research and Development, 4800 Lakewood Dr., Suite 5, Waco, TX 76710 (Modules are available individually or by course; write for prices)
EDRS PRICE	MF01 Plus Postage. PC Not Available from EDRS.
DESCRIPTORS	Learning Activities; Learning Modules; *Nuclear Energy; Postsecondary Education; Quality Control; *Technical Education; Technical Occupations
IDENTIFIERS	Nuclear Energy Occupations; Nuclear Power Plants; *Nuclear Reactors; *Nuclear Technology; Support Systems

ABSTRACT

This technical specialty course is one of thirty-five courses designed for use by two-year postsecondary institutions in five nuclear technician curriculum areas: (1) radiation protection technician, (2) nuclear instrumentation and control technician, (3) nuclear materials processing technician, (4) nuclear quality-assurance/quality-control technician, and (5) nuclear reactor (plant) operator trainee. This course, Reactor Auxiliary Systems, contains four modules: Containment and Ventilation of Reactors, Spent Nuclear Fuel Storage and Handling Procedures, Emergency Power Systems for Reactors, and Residual and Emergency Cooling Systems for Reactors. Each module follows a typical format including the following sections: introduction, module prerequisites, objectives, subject matter, exercises, materials, procedures, data tables, reference materials, and tests. (A master index and suggested program planning guides are also available--see CE 026 856 and CE 026 639-643.) (LRA)

****
* Reproductions supplied by EDRS are the best that can be made *
* from the original document. *
****

ED192149

Nuclear Technology Series

COURSE 9
REACTOR AUXILIARY SYSTEMS

TECHNICAL EDUCATION RESEARCH CENTER - SW

4201 LAKE SHORE DRIVE - SUITE 111

WACO, TEXAS 76710

CE 026 780

U S DEPARTMENT OF HEALTH, EDUCATION & WELFARE
NATIONAL INSTITUTE OF EDUCATION

THIS DOCUMENT HAS BEEN REPRODUCED EXACTLY AS RECEIVED FROM THE PERSON OR ORGANIZATION ORIGINATING IT. POINTS OF VIEW OR OPINIONS STATED DO NOT NECESSARILY REPRESENT OFFICIAL NATIONAL INSTITUTE OF EDUCATION POSITION OR POLICY

"PERMISSION TO REPRODUCE THIS MATERIAL IN MICROFICHE ONLY HAS BEEN GRANTED BY

D. M. Hull

TO THE EDUCATIONAL RESOURCES INFORMATION CENTER (ERIC)."

OCT 1980

P R E F A C E

ABOUT NUCLEAR TECHNOLOGY MODULES

The modules - developed by TERC-SW for use in two-year postsecondary technical institutions to prepare technicians for employment - are useful in industry for updating employees in company-sponsored training, and as laboratory texts in engineering schools. The principles, techniques and skills taught in the modules, based on tasks that nuclear technicians perform, were obtained from a nationwide advisory committee of employers of nuclear technicians. Each module was written by a working technical expert and tested with students.

A module contains the following elements:

Introduction, which identifies the topic and often includes a rationale for studying the material.

Module Prerequisites, which identify the material a student should be familiar with before studying the module.

Objectives, which clearly identify what the student is expected to know for satisfactory module completion. The objectives, stated in terms of action-oriented behaviors, include such action words as operate, measure, calculate, identify and define, rather than words with many interpretations, such as know, understand, learn and appreciate.

Subject Matter, which presents the background theory and techniques supportive to the objectives of the module. Subject matter is written with the technical student in mind.

Exercises, which provide practical problems to which the student can apply this new knowledge.

Materials, which identify the equipment required to complete the laboratory procedure.

Procedures, which is the experiment section, or 'hands-on' portion of the module (including step-by-step instruction) designed to reinforce student learning.

Data Tables, which are included in most modules for the first year (or basic) courses to help the student learn how to collect and organize data.

Reference Materials, which are included as suggestions for supplementary reading/viewing for the student.

Test, which measures the student's achievement of the prestated objectives.

Copyright in 1977 by the Technical Education Research Center, Inc. Printed in the United States of America. All rights reserved. These materials, or parts thereof, may not be reproduced in any form without permission.

The materials contained herein were developed pursuant to a contract with the Office of Education, U.S. Department of Health, Education and Welfare: Contract No. 300760356. Contractors undertaking such projects under Government sponsorship are encouraged to express freely their professional judgment in the conduct of the project. Points of view or opinions stated do not, therefore, necessarily represent official Office of Education position or policy.

The Nuclear Technology Series will consist of five curricula when completed:

- Nuclear Reactor (Plant) Operator Trainee.
- Radiation Protection Technician.
- Nuclear Materials Processing Technician.
- Nuclear Quality-Assurance/Quality-Control Technician.
- Nuclear Instrumentation and Control Technician.

The completed Nuclear Reactor (Plant) Operator Trainee curriculum consists of the following sixteen technical specialty courses made up of 114 instructional modules:

- Radiation Physics.
- Radiation Protection I.
- Principles of Process Instrumentation.
- Heat Transfer and Fluid Flow.
- Introduction to Nuclear Systems.
- Instrumentation and Control of Reactors and Plant Systems.
- Reactor Operations.
- Reactor Safety.
- Reactor Auxiliary Systems.
- Power Plant Systems.
- Radiation Detection and Measurement.
- Reactor Physics.
- Power Plant Chemistry.
- Introduction to Quality Assurance/Quality Control.
- Metallurgy and Metals Properties.
- Mechanical Component Characteristics and Specifications.

Availability: Instructional modules are available individually or by courses from:

Technical Education Research Center - Southwest
4201 Lake Shore Dr., Suite 111
Waco, Texas 76710
Telephone (817) 772-8756

Printed:

CONTENTS

Preface

Module 9-1 Containment and Ventilation of Reactors

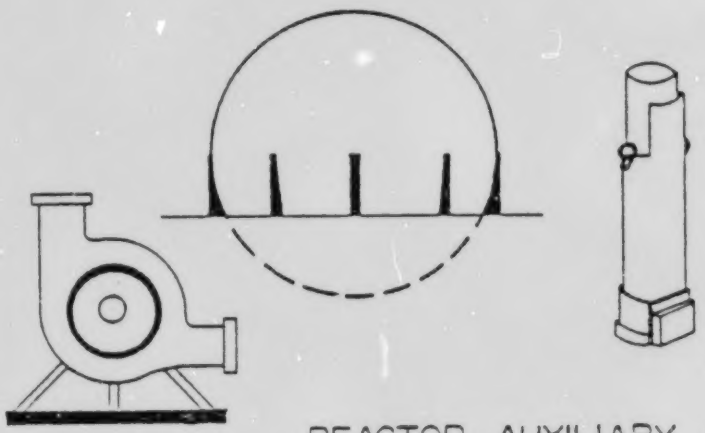
Module 9-2 Spent Nuclear Fuel Storage and Handling Procedures

Module 9-3 Emergency Power Systems for Reactors

Module 9-4 Residual and Emergency Cooling Systems for Reactors

NUCLEAR TECHNOLOGY

MODULE 9-1 CONTAINMENT AND VENTILATION OF REACTORS



REACTOR AUXILIARY SYSTEMS

COURSE 9

TERC-SW

TECHNICAL EDUCATION RESEARCH CENTER — SOUTHWEST
4201 LAKE SHORE DRIVE WACO, TEXAS

COURSE MODULES

COURSE 9 REACTOR AUXILIARY SYSTEMS

MODULE 9-1 Containment and Ventilation of Reactors
MODULE 9-2 Spent Nuclear Fuel Storage and Handling Systems
MODULE 9-3 Emergency Power Systems for Reactors
MODULE 9-4 Residual and Emergency Cooling Systems for Reactors

Copyright in 1977 by the Technical Education Research Center, Inc. Printed in the United States of America. All rights reserved. These materials, or parts thereof, may not be reproduced in any form without permission.

The materials contained herein were developed pursuant to a contract with the Office of Education, U.S. Department of Health, Education and Welfare: Contract No. OEG-0-74-1679. Contractors undertaking such projects under Government sponsorship are encouraged to express freely their professional judgment in the conduct of the project. Points of view or opinions stated do not, therefore, necessarily represent official Office of Education position or policy.

Technical Education Research Center
Southwest
4201 Lake Shore Drive, Suite 111
Waco, Texas 76710

INTRODUCTION

As knowledge of nuclear reactor technology has increased and experience in nuclear power plant operation has accumulated, emphasis has slowly shifted from primary reliance on containment structures to achievement of safety by proper reactor/plant design. While providing for the necessary containment of fission products after an accident promotes the safety of the general public, providing a design that ensures no accidents is the more proper goal.

A nuclear installation consists of three safety zones, one around the other, each of which must be breached before any significant amount of radioactive fission products is released to the surrounding area.

The cladding material, which encases the fuel, is the first protective boundary. This cladding isolates the fuel and fission products from the reactor coolant system. If fuel is not cooled properly, as during a loss-of-coolant accident (LOCA), it melts and releases gaseous and solid fission products to the primary system.

The second barrier is the reactor vessel and components of the primary system.

The final release barrier is a containment vessel or structure. This structure contains the primary coolant system, including the reactor coolant pumps, reactor vessel, steam generators, reactor coolant piping and pressurizer PWR system. In a BWR system, the structure contains the reactor vessel, piping, suppression chamber, pumps and other components. These structures are designed as pressure vessels using

standards developed by the American Society of Mechanical Engineers (ASME) and the American Concrete Institute.

These three protection areas are designed to complement each other and protect the public and the environment. The final barrier, the containment structure, is discussed in this module.

MODULE PREREQUISITES

The student should have completed a course in basic heat transfer and fluid flow, as well as having completed Course 5, "Introduction to Nuclear Systems."

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Diagram BWR and PWR containment structures, including reactors, steam generators (if applicable) and other associated facilities, such as suppression pools and ice condensers.
2. List and describe typical methods of reducing pressure (in a containment structure) following a loss-of-coolant accident.

3. Explain why the containment structure must be capable of withstanding elevated internal pressures.
4. Draw and explain typical containment structures, indicating how each is designed to suppress internal pressure from a coolant piping rupture.
5. Explain the two phases of energy release from a coolant piping rupture.
6. List and explain load considerations in the design of a containment structure.
7. Sketch and explain the three general types of containment penetration.
8. Sketch and explain a typical containment ventilation system.
9. Sketch and explain a typical containment spray system.
10. Sketch and explain operation of a system for detecting gaseous and particulate radioactive substances in the containment building.
11. List two methods of removing gaseous H_2 after a loss of coolant accident.

PWR CONTAINMENT

Most PWR systems are housed in the familiar dome-shaped building commonly associated with nuclear plant sites (Figure 9-1-1). The containment buildings house

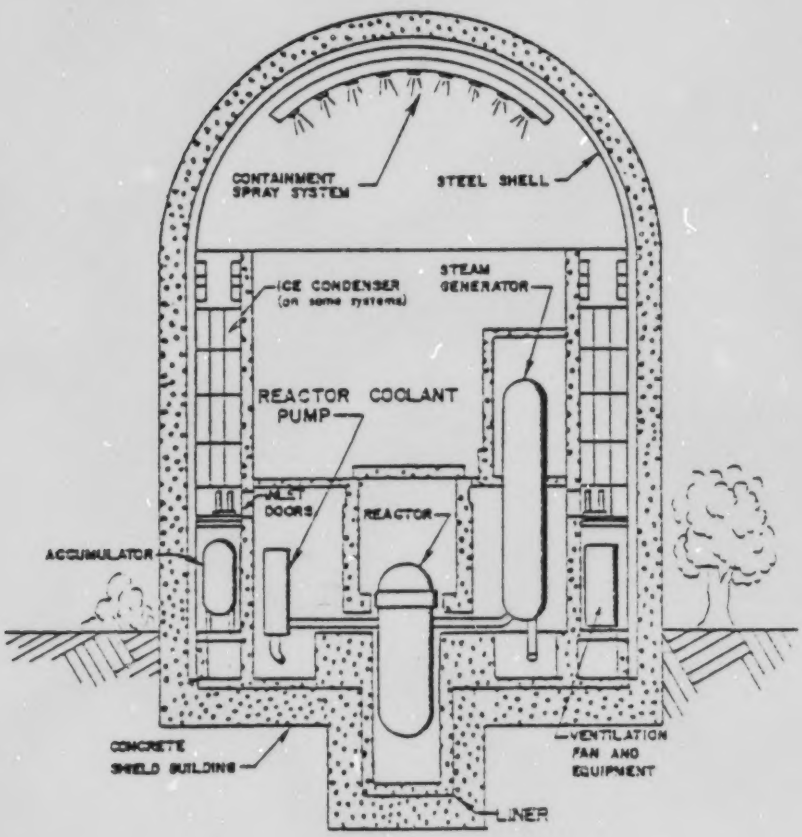


Figure 9-1-1. Typical PWR containment building.

major primary components, including reactor vessel, steam generator(s), reactor coolant pumps and pressurizer.

Containment buildings must be designed to withstand internal pressures up to 70 psia, caused by the release of high-pressure (2200 psia) water at 600°F. When this water is released to the atmosphere, as through a pipe break, it immediately flashes (vaporizes) to steam with a corresponding pressure increase to the building.

The structure is usually constructed of high-strength, reinforced concrete to contain the pressure. It may be lined on the inside with a steel membrane erected separately or in conjunction with the concrete shell.

The concrete structure is further strengthened by post-tensioned tendon cables installed in conduits that can be individually maintained and inspected. This type of construction provides greater safety along with reduced amounts of concrete used in the building.

An auxiliary reactor-building spray system sprays cooling water on the steam in the containment during an LOCA. The steam is condensed, thereby reducing pressure.

Within the containment structures are polar cranes or lifting equipment for reactor vessel head and fuel-handling operations. Containment structures are 200 to 240 feet high, measured from floor to top of dome. They incorporate an equipment hatch, personnel air locks and fuel transfer tubes (highly engineered from safety and reliability standpoints, with facile provisions for leak testing). The containment is normally closed, but must be accessible during reactor operations.

A variant PWR design, used in some plants, employs ice condenser containment; steam released as a result of pipe

rupture is condensed by the melting of a volume of ice situated within the building. The advantage of this type containment is low peak pressures, generally on the order of one-quarter less than developed in dry structures during an LOCA. It is postulated that pressure is reduced to a few pounds per square inch very rapidly after an accident, resulting in a substantial reduction in the possibility of radioactive gas leakage to the surrounding environment.

The need for auxiliary features requiring external power sources for post-accident pressure reduction is eliminated by the incorporation of this static safety feature. (It is always there and ready, and does not need to be activated.) Ice containment also allows the interior volume of the structure to be approximately one-half as large as a conventional dry containment that uses sprays.

BWR CONTAINMENT

Generally, boiling water reactor containments are of the pressure-suppression type; energy released by an accident is dissipated into water contained in a (pressure) suppression pool.

Steam and hot fluids resulting from an LOCA are discharged into a pool of water. The energy in the hot fluid is then dissipated into the cold fluid, greatly reducing containment volume requirements. The water also entraps radioactive materials, reducing any potential hazard to the public.

The original BWR Mark I containment system (Figure 9-1-2) looked like an inverted light bulb connected to a doughnut-shaped suppression chamber by a series of pipes. The suppression chamber contains a pool of water to condense any energy release in the vessel.

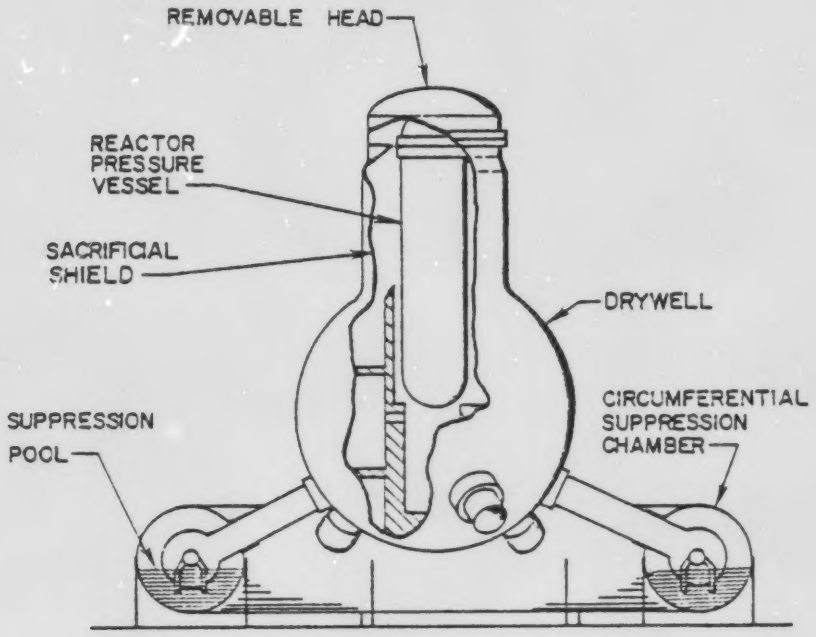


Figure 9-1-2. Mark I containment design. (General Electric Co.)

The containment volume of this design is much smaller than the dry-well containment of a PWR reactor of comparable output capacity. This reduction in size is possible because of the condensing function in the suppression chamber.

A second type of containment, also designed by the General Electric Company for BWR systems, is the Mark II (Figure 9-1-3). It differs significantly from the Mark I

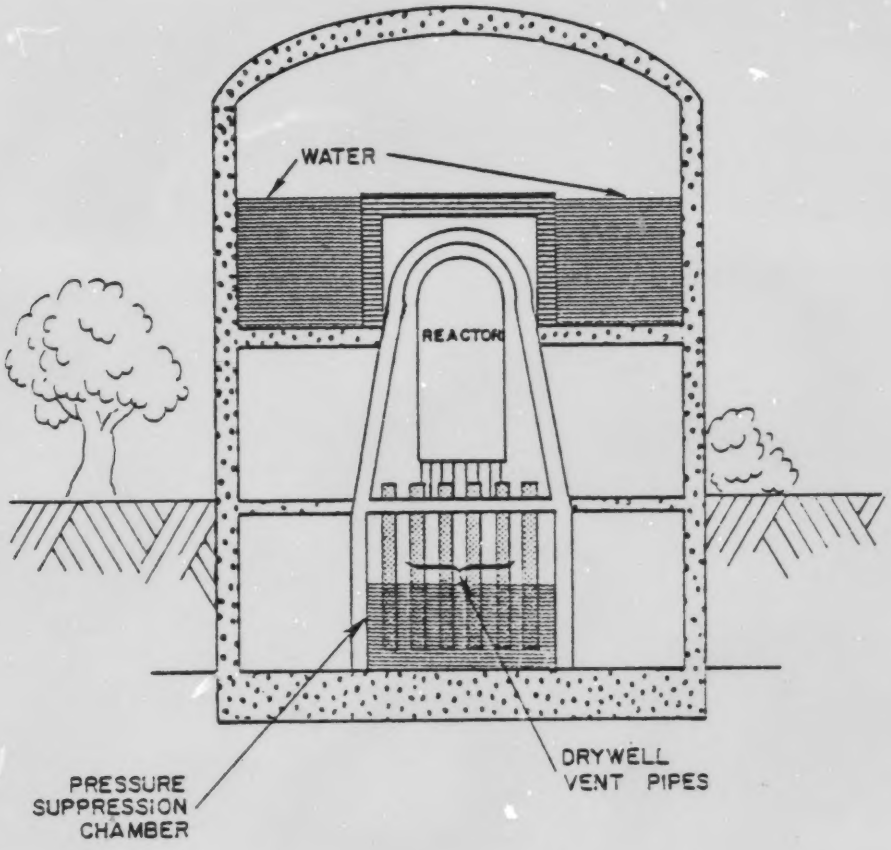


Figure 9-1-3. Mark II containment system design.

in that it is shaped like an inverted cone mounted on a cylindrical base, with the reactor situated in the upper cone section. The bottom of the structure contains the pressure-suppression pool, is more structurally sound and generally more desirable in terms of cost for a given safety characteristic.

A later design, the Mark III (Figure 9-1-4), combines features of a dry-containment structure with the pressure-suppression features of the Mark I and Mark II designs; thus, it has an even greater safety margin through its multiplicative suppressor arrangement.

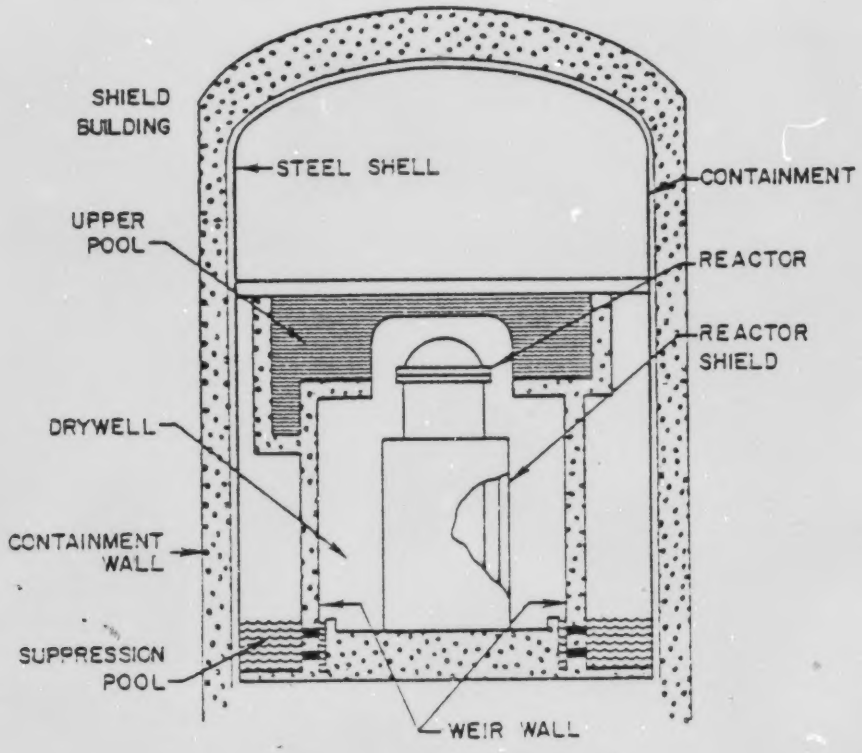


Figure 9-1-4. Mark III containment design.

In contrast to most PWR containments, the dry-well structure of a Mark III does not have a metal membrane - it consists of reinforced concrete only. Its function also includes shielded access to the building during operation, and support for ancillary equipment.

The suppression pool in this design is between the weir wall and the containment wall. This arrangement allows simplified piping runs from the dry well to the suppression pools. The rectangular structure in the upper part of the dry well provides space for refueling operations. The pool, which is filled with water during reactor operation, serves as a shield to further reduce levels during normal plant operation.

BASIS OF DESIGN CONDITIONS

The containment structure is the principal barrier to release of radioactive substances to the environment; therefore, the design basis is premised on the most improbable event that could occur (for example, a simultaneous earthquake and LOCA). The design goal is to ensure safe plant shutdown under these hypothetical, highly unlikely circumstances.

An LOCA is any condition in which a rupture in the primary coolant piping releases massive amounts of energy into the containment building. It is characterized by two phases of energy releases.

Phase I is the "blowdown." High-pressure coolant water discharges from the break, and a portion flashes

into steam (because of the comparatively low pressure within the containment). High discharge rates during blowdown relieve the reactor coolant pressure very rapidly.

Phase II is the post-blowdown phase. Following blowdown, additional heat transfers to the containment. This heat includes decay heat, heat stored in reactor components (such as internal core energy), reactor vessel metal energy, steam generator energy, metal water reaction energy and any coolant energy. This post-blowdown phase is characterized by a thermal transient in which thermal discharge rates depend upon core flooding rates.

Additional engineering safety features must be provided to satisfy Title 10, Code of Federal Regulations, Part 100, as promulgated by the Nuclear Regulatory Commission (NRC).

In addition, design consideration for steel structures must comply with the latest ASME Boiler and Pressure Vessel Code. And those using concrete are regulated by building code requirements of the American Concrete Institute. In addition, these buildings must also comply with applicable local building codes.

The design of the structure must specifically take into consideration:

Dead Loads, defined as equipment loads, loads provided when equipment is installed, pre-stressed loads and temporary loads applied during construction.

Live Loads, which are variable loads, including those resulting from pipe systems or other outside dynamic loads.

Environmental Loads, which include wind, tornadoes, earthquakes and buoyant forces.

DESIGN-BASIS ACCIDENT

Containment structures must be able to withstand forces generated by a "design-basis accident," as defined in General Design Criteria 16 and 50, Appendix A, 10CFR50, which reads as follows:

> Criterion 16 - Containment design. Reactor containment and associated systems shall be provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.
>
> Criterion 50 - Containment design basis. The reactor containment structure, including access openings, penetrations and the containment heat removal system, shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss of coolant accident. This margin shall reflect consideration of:
>
> The effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result

from degraded emergency core cooling functioning;

The limited experience and experimental data available for defining accident phenomena and containment responses; and

The conservatism of the calculational model and input parameters.

CONSTRUCTION

Containment construction differs from ordinary construction in many ways. A stringent quality-assurance program must be adhered to continually during the construction period. This program includes records of materials used, tests for integrity, and constant surveillance by NRC inspectors.

Various methods of erecting the liners and reinforced concrete containment walls may be employed. The liner and concrete may be erected simultaneously or the steel membrane may be emplaced and the concrete formed around it (called slip forming).

ACCESS AND PENETRATIONS

Access for personnel and equipment into the containment building must be carefully considered to ensure integrity of the structure insofar as containment and positive

sealing criteria are concerned. Personnel must have access through the entrance opening at all times; therefore, openings are usually air locks (Figure 9-1-5).

The personnel air lock entrance must be a coupled, double-door arrangement; when one door is closed the other is open. Because of the weight of these doors, they must be counterbalanced to minimize effort required for opening. Also, the access door must adhere to sealing requirements and design codes for containment building penetrations.

Equipment hatches are also installed in most structures. These hatches, used only during shutdown conditions, are sized to permit entry of the largest pieces of expendable and maintenance equipment. The equipment hatch and two doors associated with the air lock all have double gaskets with provisions for testing leaks in the space between them.

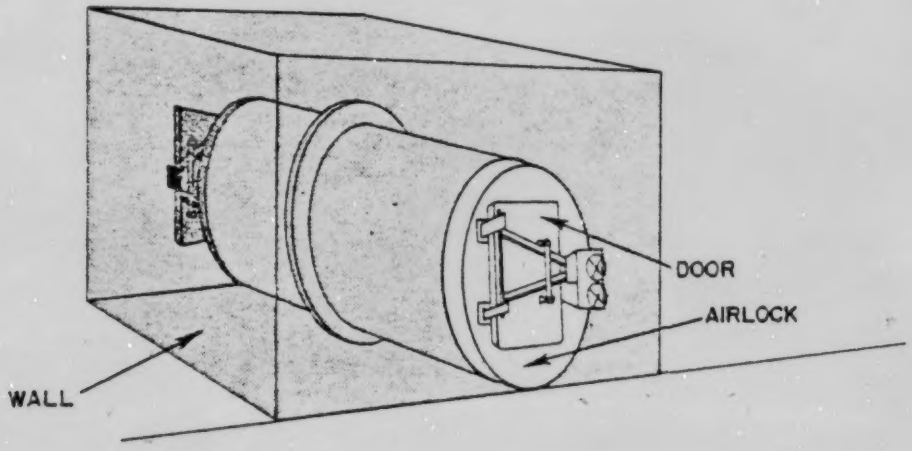


Figure 9-1-5. Personnel access door (air lock).

Also, mechanical penetrations are provided for all piping lines to or from the containment building. These penetrations are divided into two main thermal classifications:

Hot penetrations.
Cold penetrations.

Both hot and cold penetrations must be sealed and anchored to the containment wall, and must not be overstressed by the thermal- or seismic-induced motion. Furthermore, they must withstand normal mechanical loads, while minimizing any stressing of the containment shell. Any penetration passing fluids that exceed 200°F is normally considered to be a hot penetration and must be provided with cooling capability to prevent thermal stressing of the containment structure. This requirement is met by one of the auxiliary cooling systems.

Electrical penetrations (classified as power cable penetrations or instrument cable penetrations) are classified as "cold" (Figure 9-1-6). The design of these penetrations must provide for leak tightness and integrity during both normal and abnormal conditions.

Often a penetration may have multiple cables within its boundary, in which case care must be taken that redundant equipment is separated into different penetration paths.

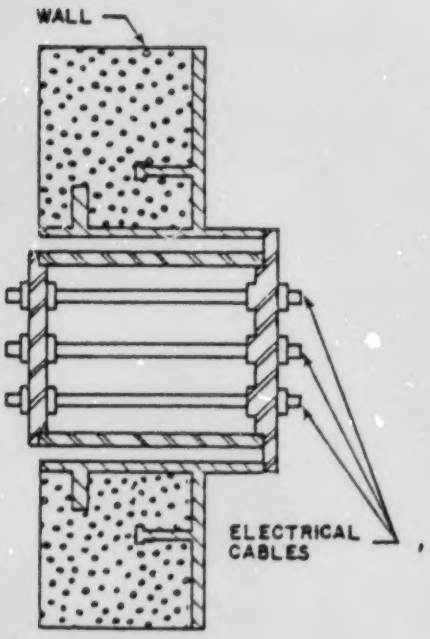


Figure 9-1-6. Typical "cold" penetrations.

HEAT REMOVAL AND VENTILATION

To ensure containment integrity in the event of an LOCA or steam line break, a heat removal system is necessary in both PWR and BWR systems. These removal systems are comprised of both active and passive components.

RADIATION MONITORING OF CONTAINMENT VENTILATION

Radiation monitoring of containment ventilation is provided for in the detection of reactor coolant leakage. A sample volume of reactor containment air is drawn through a set of radiation detectors that determine particulate and gaseous activity. These radiation detectors (Figure 9-1-7) are sufficiently sensitive to provide detection of small reactor coolant leaks within thirty minutes.

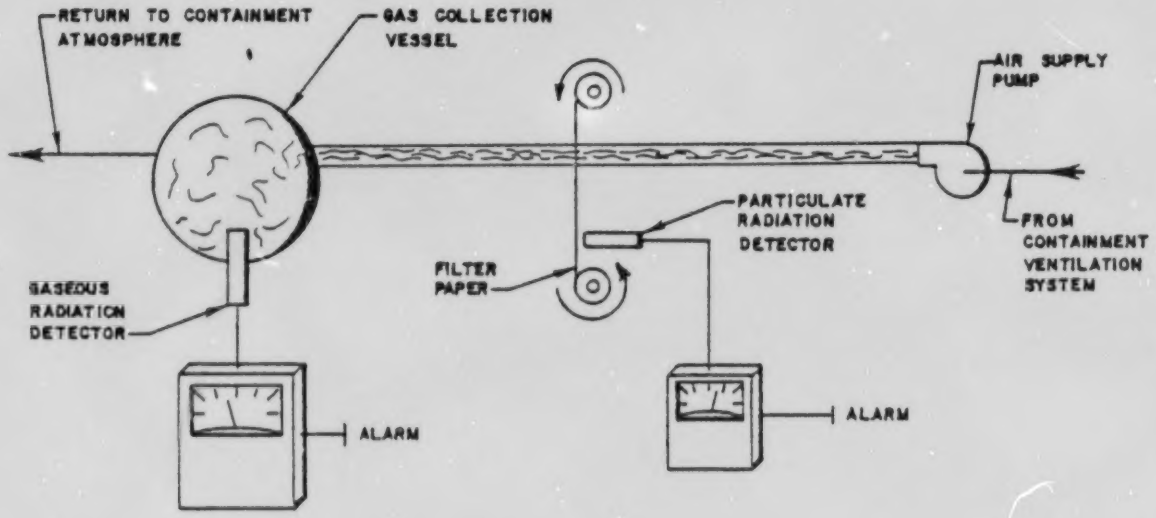


Figure 9-1-7. Radiation monitor system.

The sample is pumped through continuously moving particulate filter paper. The paper is then passed in front of a radiation detector. Radioactive particles trapped on the filter are detected by the monitor.

The filtered air passes into a chamber where any residual gaseous activity is determined by radiation detection. The output of this detector provides indication, alarm and control functions for the operator.

POST-ACCIDENT H_2 REMOVAL

It is postulated that, during an LOCA, a large volume of hydrogen gas could be generated by the reaction of metals with the high-temperature water. This hydrogen could become an explosive hazard; therefore, it must be removed or diluted to permissible levels by a post-accident H_2 removal system.

This system can be one of two types. The first is post-accident hydrogen dilution, accomplished by 1) controlled release of a quantity of the containment atmosphere to the atmosphere through high-efficiency filters and 2) dilution of the H_2 gas by the introduction of clean air into the containment.

The second is a hydrogen-recombination system; hydrogen gas is recombined with oxygen in the presence of a catalyst to form water under nonexplosive conditions.

EXERCISES

1. List the three safety zones of a nuclear system.
2. What major components are housed in a PWR containment building? In a BWR containment building?
3. How does the use of suppression-pool or ice-condenser containment affect the size and pressure requirements of a containment building?
4. What materials are normally used in reactor containment building construction?
5. Why must all mechanical penetrations be sealed and anchored to the containment wall?
6. List two ways to reduce the possibility of an H_2 explosion after an LOCA.

MATERIALS

None.

PROCEDURES

Visit a PWR site and a BWR site and:

1. Observe type of containment building.
2. Study access door seal arrangement.

3. Observe heat removal and ventilating system.
4. Check containment ventilation monitoring system.

DATA TABLE

None.

REFERENCE MATERIALS

American Society of Mechanical Engineers. ASME Boiler and Pressure Vessel Code. New York: ASME, 1974.

Babcock & Wilcox Co. Steam. 38th ed. New York: B & W, 1976.

Combustion Engineering, Inc. System 80. Windsor, CT: Combustion Engineering, Inc., 1976.

Miller, Franklin, Jr. College Physics. 3rd ed. New York: Harcourt Brace Jovanich, Inc.

United States Nuclear Regulatory Commission. "General Design Criteria for Nuclear Plants." Title 10, Code of Federal Regulations, Part 50, Appendix A.

Available by writing the Superintendent of Documents, U.S. Government Printing Office, Washington, DC 20402.

1. Sketch each of the three most common designs of containment buildings associated with PWR systems (use a separate sheet):
2. Explain why a containment structure must be capable of withstanding elevated internal pressures:

3. Briefly describe the two most typical methods of pressure suppression in a PWR containment structure:

4. Sketch the three most typical BWR containment structures; explain how each is designed to suppress energy released in a coolant piping rupture (use separate sheet):
5. List and briefly explain the two general phases of energy release in a coolant piping rupture:

6. Name and explain the three general types of loads taken into account in the design of a containment structure:

7. Briefly explain the substance of Criteria 16 and 50 in Appendix A to 10CFR50:

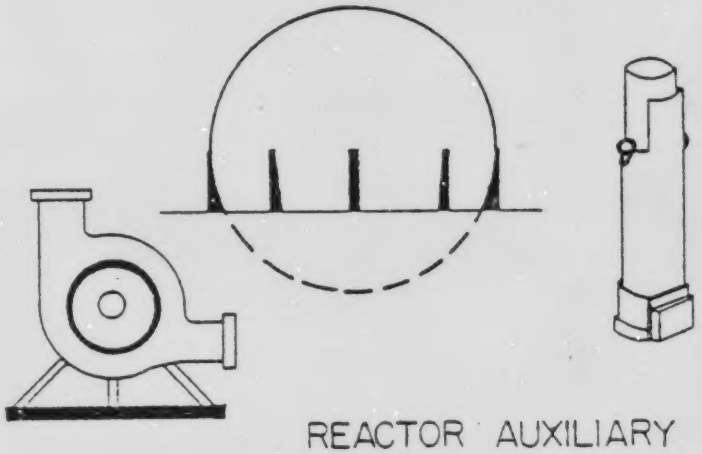
8. Sketch (on a separate sheet) and explain each of the following:
   An electrical penetration.
   A piping penetration.
   A personnel air lock.
9. Sketch and briefly explain a typical containment structure cooling/ventilation system (use a separate sheet):
10. Sketch and explain a basic containment spray system:

11. Sketch (on a separate sheet) and explain a simple system for detection of gaseous and particulate radioactive substances in a containment structure:

12. List and briefly explain two methods of H_2 removal:

NUCLEAR TECHNOLOGY

MODULE 9-2 SPENT NUCLEAR FUEL STORAGE AND HANDLING PROCEDURES



REACTOR AUXILIARY SYSTEMS

COURSE 9

TERC-SW

TECHNICAL EDUCATION RESEARCH CENTER — SOUTHWEST
4201 LAKE SHORE DRIVE WACO, TEXAS

COURSE MODULES

COURSE 9. REACTOR AUXILIARY SYSTEMS

MODULE 9-1 Containment and Ventilation of Reactors

MODULE 9-2 Spent Nuclear Fuel Storage and Handling Procedures

MODULE 9-3 Emergency Power Systems for Reactors

MODULE 9-4 Residual and Emergency Cooling Systems for Reactors

Copyright in 1977 by the Technical Education Research Center, Inc. Printed in the United States of America. All rights reserved. These materials, or parts thereof, may not be reproduced in any form without permission.

The materials contained herein were developed pursuant to a contract with the Office of Education, U.S. Department of Health, Education and Welfare: Contract No. 300760356. Contractors undertaking such projects under Government sponsorship are encouraged to express freely their professional judgment in the conduct of the project. Points of view or opinions stated do not, therefore, necessarily represent official Office of Education position or policy.

Technical Education Research Center
Southwest
4201 Lake Shore Drive, Suite 111
Waco, Texas 76710

INTRODUCTION

A fuel-handling system provides for performance of all fuel-handling operations (from initial fuel loading to re-fueling) required during plant lifetime. This system ensures safe as well as effective handling of fuel assemblies and control-rod assemblies from the time of their arrival on site until off-site shipment occurs. Components within the system are designed and constructed to minimize: 1) time required for fuel handling and 2) possibility of damage to fuel.

MODULE PREREQUISITES

The student should have completed Course 2, "Radiation Protection," Course 5, "Introduction to Nuclear Systems," Course 6, "Instrumentation and Control," Course 11, "Radiation Detection and Measurement," and Course 12, "Reactor Physics."

OBJECTIVES

Upon completion of this module, the student should be able to:

1. List four purposes of a fuel-handling system.
2. Describe journey of a new fuel assembly, from arrival at facility, through use in core, to transfer to a fuel reprocessing plant.
3. Sketch and explain the multiple-function mast associated with a main fuel-handling bridge.
4. List and explain functions of the four main fuel-handling bridge-limiting circuits.
5. Sketch and explain operation of a typical fuel-transfer system.
6. List and explain carriage and rotating-frame interlocks associated with a fuel-transfer system.
7. Explain benefits to be gained by using an auxiliary fuel-handling bridge.
8. List and discuss considerations included in planning and scheduling a refueling period.
9. Describe special heat and physics problems associated with refueling.

FUEL-HANDLING SYSTEM

A fuel-handling system (Figure 9-2-1) is designed to perform all handling operations under water. During a fuel-handling operation, the system provides for the following:

Transferring of spent fuel assemblies from reactor vessel to fuel-storage pool.

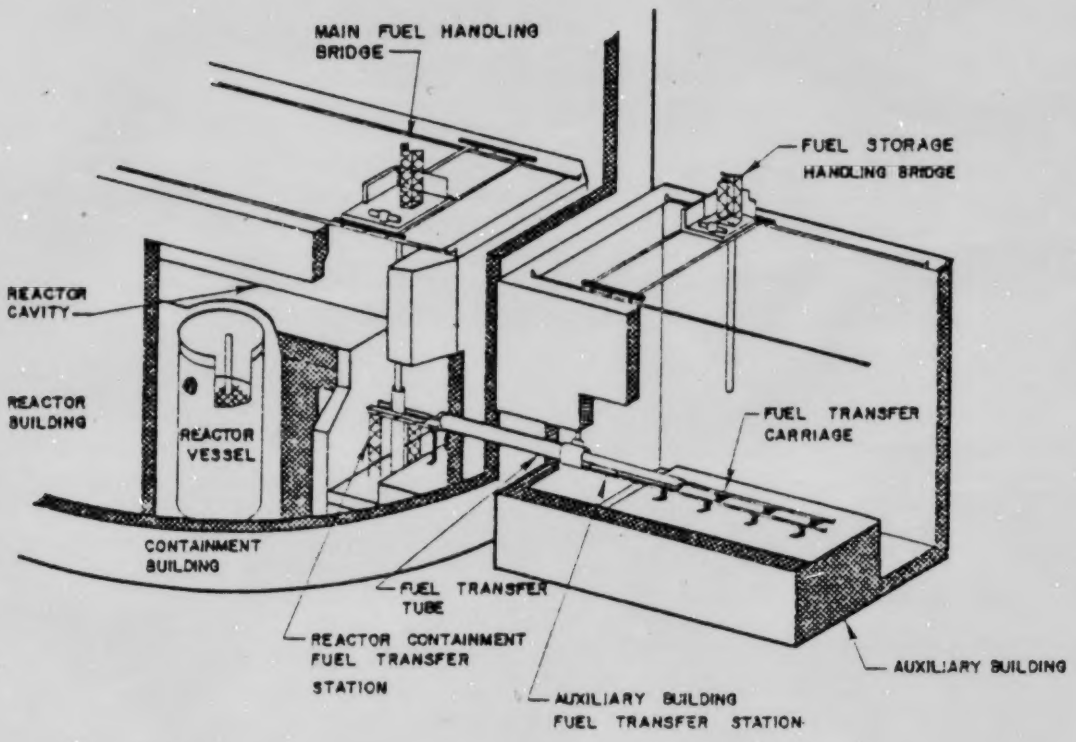


Figure 9-2-1. Fuel-handling system.

Relocation of partially spent fuel assemblies from one station within the core to another.

Transfer of new fuel assemblies from storage area into core.

Handling of control rods, axial power-shaping rods, burnable poison rods and orifice rod assemblies, as required, to place these control components in their proper location in reactor core.

REACTOR CAVITY AND REFUELING CANAL

The reactor vessel is located in a well, called a "reactor cavity," in the reactor building. The reactor cavity, an integral part of the reactor building, is constructed of reinforced concrete and lined with stainless steel.

During normal operations, the cavity houses reactor, reactor head and service structure. During refueling operations, a seal is placed between reactor vessel flange and cavity wall. The cavity, which is filled with borated water to provide radiation shielding and cooling for the irradiated fuel, is then called the "refueling canal."

The refueling canal provides storage for upper components, which must be removed to expose fuel assemblies. These components are highly radioactive and must be shielded by water in the reactor cavity.

In preparation for fuel handling, reactor cavity is filled, and upper internal components are removed and set aside in reactor cavity; also, communication is established between reactor cavity and fuel storage area. During fuel-handling operations, fuel-handling equipment performs all distinct sequential operations necessary to complete a fuel

assembly removal and replacement (handling) cycle.

A spent fuel assembly is removed from the core and transported to a reactor containment fuel-transfer station by way of a main fuel-handling bridge. It is loaded into a fuel-transfer carriage, which will transport it through a fuel-transfer tube into an auxiliary building. Once a fuel assembly arrives at the auxiliary building fuel-transfer station, the bridge removes it from the carriage and takes it to a designated fuel-storage rack for deposit.

A new fuel assembly is transported from its designated storage rack location to its new location in the core in a reverse sequence.

When main fuel-handling bridge is not being used to transfer fuel assemblies to or from fuel-transfer station, it can be used to shuffle fuel assemblies within the core to equalize fuel burnup.

Installation of a second, or auxiliary, fuel-handling bridge within the reactor containment building allows fuel shuffling to take place during fuel transfer.

MAIN FUEL-HANDLING BRIDGE

The main fuel-handling bridge (Figure 9-2-2), situated in the reactor containment building, is equipped with a multiple-function mast (Figure 9-2-3) designed to handle fuel assemblies or control rods during fuel-handling operations. A multiple-function mast consists of a fixed outer mast, an inner mast, fuel- and control-rod grapples and systems for raising and lowering fuel assemblies and control rods.

The fixed outer mast houses (and provides guidance to ensure vertical alignment of) the inner mast assembly. Dur-

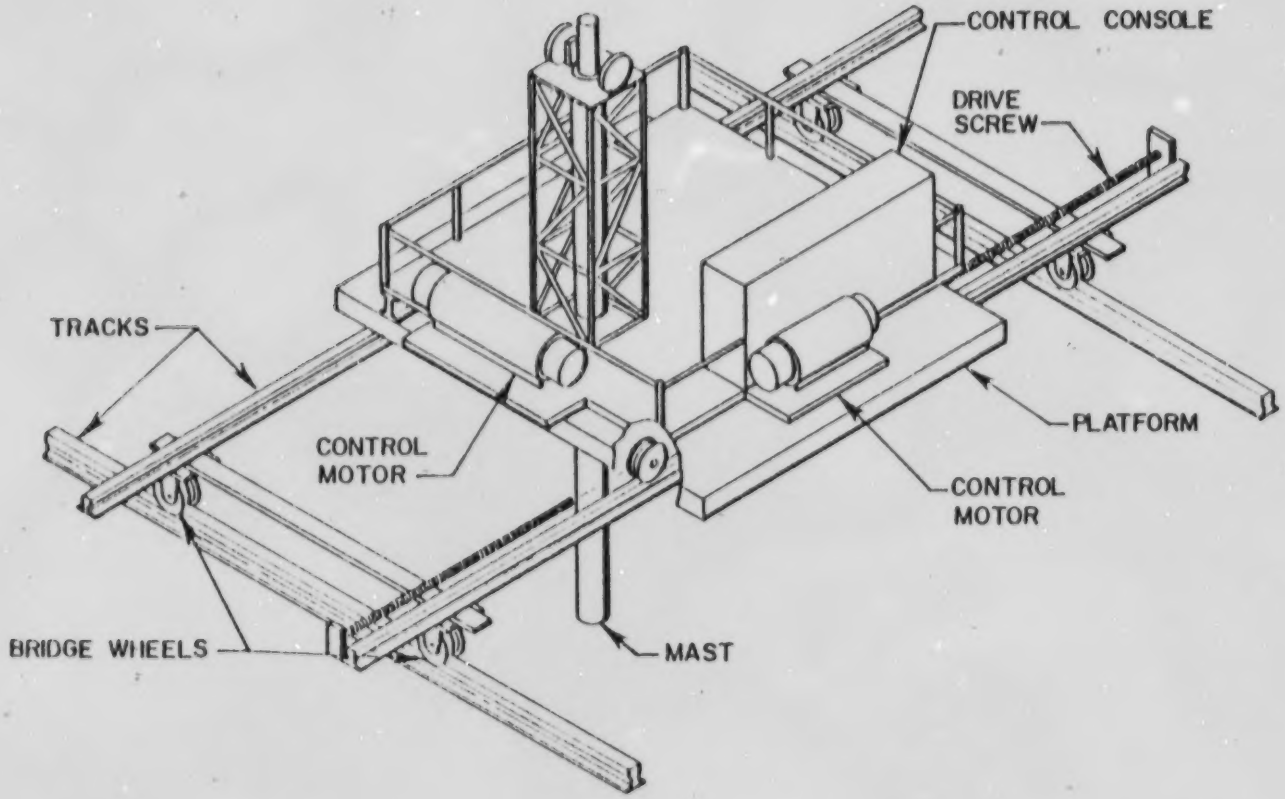


Figure 9-2-2. Main fuel-handling bridge.

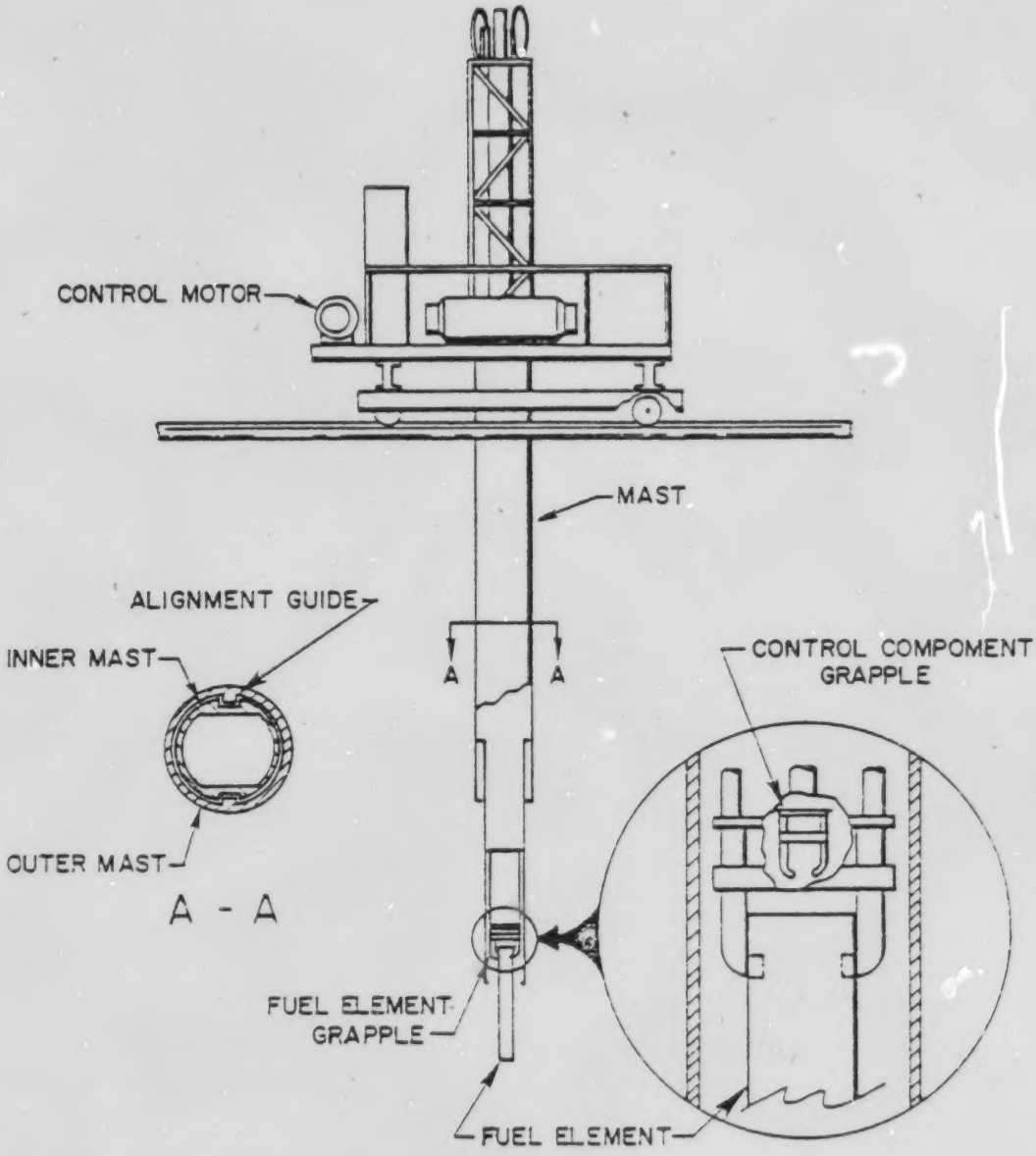


Figure 9-2-3. Main fuel-handling bridge with multiple-function mast.

ing fuel-handling operations, when the inner mast assembly is at its up position and the fuel grapple is engaged with a fuel assembly, the fixed outer mast houses and protects fuel assemblies throughout bridge and trolley transport movements.

The inner mast assembly has a fuel-grapple assembly attached to its lower end; thus, it incorporates alignment guides to provide positioning of fuel grapple for engagement with fuel assembly. During vertical travel, vertical alignment is maintained by inner mast alignment guides situated in the fixed outer mast.

Also housed within the inner mast assembly are control-rod handling guides and control-rod grapple, which move up and down within the inner mast. Alignment of control-rod grapple with control component is accomplished by the grapple engaging the control component coupling. After engagement, the control-rod grapple is raised, withdrawing the control component from the fuel assembly, up through the disengaged fuel grapple until the control component is housed completely within the inner mast assembly. The inner mast assembly is then raised to its "up" position within the fixed outer mast.

One hoist, attached to the inner mast assembly via a mechanical linkage from the control-rod grapple, provides for all bridge raising and lowering functions. The hoist usually has two-speed circuitry. A slow-speed circuit is included as a safety feature to be used when inserting or withdrawing a fuel assembly or control component - the possibility exists of binding an element. During all other handling operations, hoist operation proceeds at maximum speed.

A hoist load-monitoring system continuously measures and displays the loads supported by grapples. In this sys-

tem are the limiting circuits for low-load, overload, fuel-grapple electrical interlock and underload. Hoist load circuit functions include:

Low-load (slack cable) circuit stops the hoist automatically in down direction when component being handled and its weight are no longer supported by grapple.

Overload circuit stops hoist automatically in up direction any time design load limit is exceeded. An excessive load is indicative of a stuck fuel assembly or control component.

Fuel-grapple electrical-interlock circuit prevents disengagement of grapple when they are supporting weight of handled component; to allow disengagement could result in dropping and damaging a fuel assembly or control component. Manual disengagement of fuel- or control-rod grapple is permitted only when element has been lowered into position and weight is no longer supported by grapple.

Underload circuit stops the hoist automatically in down direction when load being handled decreases below a preset limit. While lowering fuel assembly into position, circuit is in effect until approximately the last one inch of downward travel, at which time it is locked out of hoist control circuitry. Low-load circuit controls remaining downward hoist travel. (When underload circuit is locked out of hoist control circuitry, bottom guides of fuel assembly are engaged and guided by fuel-assembly supports in core and storage racks.)

Positive stops are located near upper end of outer mast assembly to prevent an irradiated fuel assembly or control

component from being raised above the minimum permissible depth from pool surface.

An operator control console provides indicating lights and operating controls. The indicating lights, which provide operator with visual information about operational status of equipment, are color coded for ease of reference.

Bridge positioning is accomplished by visual observation of a precalibrated, selsyn-operated dial with hour and minute hand position readout. Trolley positioning is by visual observation of a pointer moving along a fixed scale with precision position markings. These components make up a position-indicating system, which is used to index bridge and trolley to their coordinate axes positions, with respect to the following:

Fuel-assembly core locations.
Transfer station.
Storage-rack locations.
Fuel and control-rod grapple test stations.
Failed-fuel detection container location.

This system is capable of positioning the center of the fuel or control-rod grapple within required tolerance above center of fuel assembly, or control component designated for handling.

Vertical position indication is provided by observation of a numbered tape attached to inner mast assembly. Two tapes are used to indicate multiple-function mast vertical position. One tape is attached to the inner mast assembly to indicate fuel assembly elevation, and another is attached to the control-rod grapple to indicate control-rod elevation. The tapes, mounted over the operator control console, provide a continuous display of elevation of component being handled.

In the event of a power failure, bridge, trolley and

hoist functions can be carried on manually.

FUEL-STORAGE HANDLING BRIDGE

The fuel-storage handling bridge, situated in the fuel-storage building, has a single-function mast and operates over the spent-fuel storage pit. The fuel-storage handling bridge, identical to the main fuel-handling bridge, except that it is "single function," is not equipped to handle control components.

FUEL-TRANSFER SYSTEM

The fuel-transfer system (Figure 9-2-4), used during

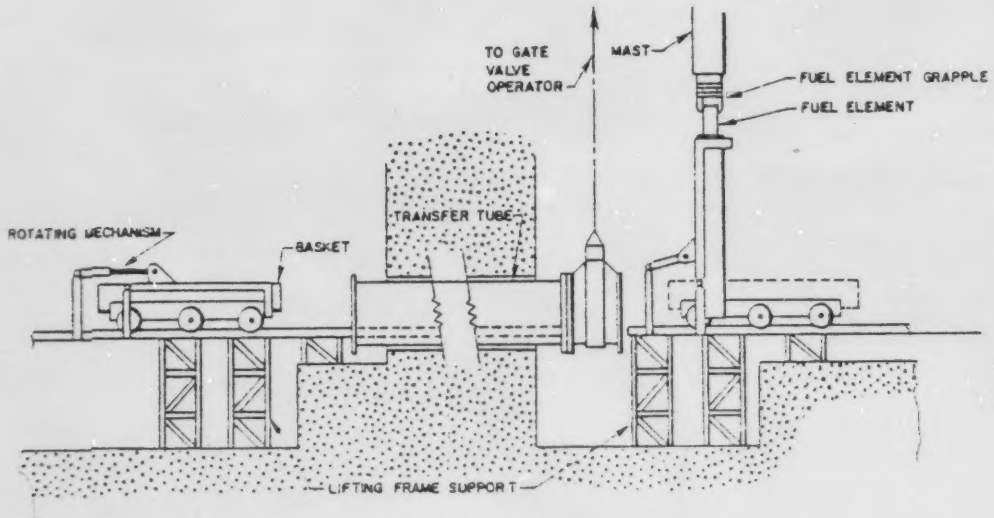


Figure 9-2-4. Fuel-transfer system.

refueling, transports fuel assemblies and control components between the auxiliary building and reactor containment building. It consists of the following components:

Two fuel-basket rotating frames.
One self-propelled carriage and fuel-basket assembly.
One fuel-transfer tube.
One manually operated gate valve, in the auxiliary building.
Two control consoles, in the reactor containment and auxiliary buildings, respectively.

During system operation, the self-propelled carriage and fuel basket assembly approach a fuel-basket rotating frame; the fuel basket's "upending guides" engage rotating-frame lifting lugs. After the carriage travels to its end limit and stops, a hydraulic cylinder is actuated, which rotates the frame and engaged fuel basket to a vertical attitude. The fuel-handling bridge is then indexed over the vertical fuel basket for removal or insertion of a fuel assembly. All handling operations can be performed with or without a control component in the fuel assembly.

Carriage and Fuel-Basket Assembly

The self-propelled carriage and fuel-basket assembly, which has a permanently attached pullout cable, is powered by an underwater air motor. The motor's supply and exhaust hoses and emergency pullout cable are assembled on reels (in the auxiliary building). These reels are spring-motor driven and automatically unwind and retract as the carriage moves back and forth between containment and auxiliary buildings. Should an air-hose fail, the emergency pullout cable manually returns the carriage to the auxiliary building.

Transfer System Control Panels

A control panel situated in the containment operates the fuel basket rotating frame in the reactor containment; a control panel situated in the auxiliary building controls the fuel basket rotating frame in the auxiliary building and horizontal travel of the carriage. Carriage and rotating frames are interlocked to prevent inadvertent movement of the carriage when:

Fuel basket is in the vertical position.
Handling bridge fuel or control-rod grapple are not required preset positions.
Manually operated transfer-tube gate valve is not in full-open position.

Fuel-Transfer Tube

A fuel-transfer tube serves as a passageway to transfer-fuel assemblies and control components, between reactor containment and auxiliary building. A gasketed cover plate, installed on reactor containment end of tube during reactor operation, is removed prior to flooding of reactor cavity. A manually operated gate valve, in the auxiliary building, is closed during reactor operation and opened during refueling.

SUPPLEMENTARY EQUIPMENT TO REDUCE FUEL-HANDLING TIME

To decrease fuel-handling time, an auxiliary fuel-handling bridge and a second fuel-transfer system and fuel-transfer tube assembly may be installed.

REFUEL TRANSFER METHODS AND SYSTEMS

New fuel is received on site in a shipping container. Each contains two fuel elements secured to strong backs in the container. Fuel is shipped in horizontal position. But, when container is opened, fuel is raised to a vertical position.

New fuel is stored dry until ready to be installed in reactor. It is removed from its shipping container, inspected and placed in a new-fuel storage rack, designed to provide proper geometry to prevent criticality in the event of accidental flooding in the storage area. Additional protection is provided by oversize drains. Fuel remains in storage until it is transferred to the fuel-storage pool via elevator (Figure 9-2-5).

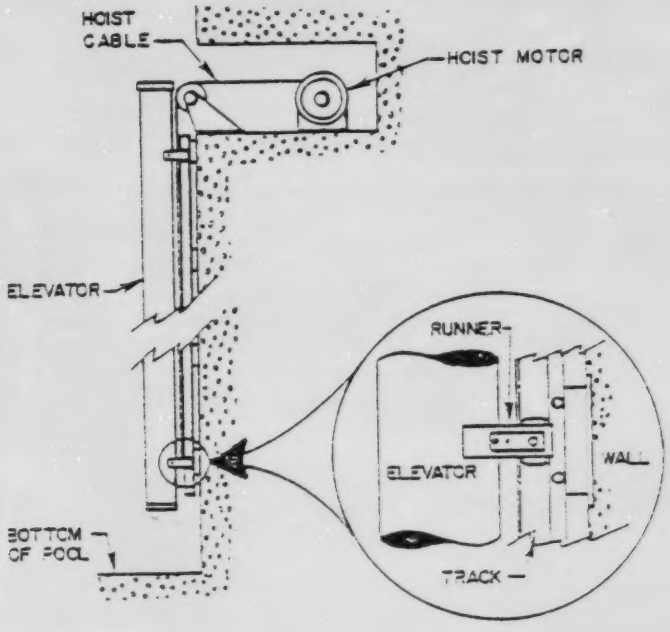


Figure 9-2-5. Fuel-handling elevator.

From the time new fuel is manufactured until it is transferred to a fuel reprocessing plant, careful records must be kept so that, at any given time, its location, exposure and storage may be determined.

REFUELING PROCEDURE - TYPICAL SCHEDULE

Long before a reactor is shut down for refueling, refueling work begins with planning and scheduling. An order for replacement fuel must be placed well in advance to allow for its manufacture and delivery, unloading, inspection and transfer into the fuel-storage pool.

When a reactor is shut down for refueling, check-out of the fuel-handling equipment begins. While the reactor is being cooled down, preparation for fuel-handling operation is made in the containment. Tools and equipment are prepared and tested.

When the reactor has been cooled down, degassed and depressurized, a seal is placed between the reactor flange and reactor cavity wall. The reactor head is unbolted, and control-rod drives are disconnected from the control rods. When all attachments have been disconnected from the reactor, guide studs are installed and the reactor head is removed and placed on a storage stand. The fuel-transfer canal is flooded with borated water to provide shielding and cooling for the fuel. Upper reactor internals are removed and placed on storage stands in the fuel canal.

Removal of the upper internals package exposes the fuel for handling. Fuel elements are then removed and transferred to the storage area, where they remain until shipped off site for reprocessing.

During this time, spent fuel is transferred, partially used fuel is relocated in the core, and control components

are shifted into their proper core positions. While spent fuel is moved to the storage pool, new fuel is transferred into the reactor building and placed into the reactor.

Once the final fuel element is placed in the reactor, the process of "closing" begins: upper internals are replaced into the reactor; the fuel transfer canal is drained; the head is replaced, and control-rod drives are connected to control rods; the reactor coolant system is closed, filled, vented and hydrostatically tested; and preparations for reactor startup and testing are made.

When planning and scheduling a reactor refueling, several things must be considered:

Work location. How many areas are available for working at a time?

Workforce. How many crews are required? How many personnel per crew are needed? What skills are needed in each crew?

Priority. When may each job be started and when must it be completed to allow other jobs to proceed on schedule?

HEALTH PHYSICS PROCEDURES

In addition to normal health physics procedures required in a nuclear power plant, additional requirements arise during refueling operations.

Irradiated fuel presents a very important problem in that it is highly radioactive and must be kept completely submerged to shield operators from its radiation. As a re-

sult, caution and safety circuits must work together to keep fuel at a safe distance below water surfaces.

Damaged fuel assemblies also create substantial problems; fission products released are highly radioactive and can result in high radiation levels near the pool surface. Also, gaseous isotopes can become airborne, creating an increased area of radiation level.

The greatest danger arises, not from potential radiation during refueling, but from failure to adequately anticipate and prepare for radiation problems in advance.

Additional personnel monitors may be required on wrists and ankles when extremity exposure is expected.

With these general health physics precautions, a safe refueling may be accomplished; however, if work on partially spent fuel is required, other special health physics precautions may be required.

EXERCISES

1. What is a reactor cavity?
2. How is a reactor cavity constructed?
3. What components are located in a reactor cavity during reactor operation?
4. How is a reactor cavity converted into a refueling canal in preparation for refueling?
5. List, in sequential order, operations performed by fuel-handling equipment.

6. What parts make up a multiple-function mast (in a main fuel-handling bridge)?
7. List interlocks on fuel-handling equipment.
8. What is the function of a bypass switch? When is it used?
9. Briefly describe receipt and handling of new fuel.
10. List, in order of their performance, steps required to refuel a reactor, from reactor shutdown to reactor start-up.

MATERIALS

None.

PROCEDURES

None.

DATA TABLE

None.

REFERENCE MATERIALS

Asimov, Isaac. Inside the Atom. New York: Abelard-Schuman Ltd., 1974.

Glasstone, Samuel. Sourcebook on Atomic Energy. 3rd ed. New York: D. Van Nostrand Co., Inc., 1967.

Nuclear Power and the Environment: Questions and Answers. New York: Unipub, 1973.

1. List four purposes of a fuel-handling system:

2. Briefly describe the journey of a new-fuel assembly, from arrival at facility to use in core to transfer to a fuel-reprocessing plant:

3. Sketch and briefly explain a multiple-function mast (associated with a main fuel-handling bridge):

4. List and briefly explain functions of four main fuel-handling bridge limiting circuits:

5. Sketch and briefly explain operation of a typical fuel-transfer system:

6. List and briefly explain a carriage and a rotating frame interlock (associated with fuel-transfer system):

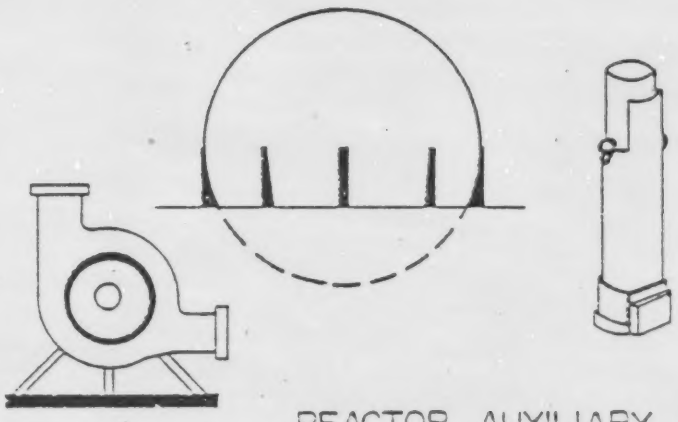
7. Briefly explain benefits to be gained from using an auxiliary fuel-handling bridge:

8. List considerations in planning and scheduling a refueling period:

9. Briefly describe special heat and physics problems associated with refueling:

NUCLEAR TECHNOLOGY

MODULE 9-3 EMERGENCY POWER SYSTEMS FOR REACTORS



REACTOR AUXILIARY SYSTEMS

COURSE 9

TERC-SW

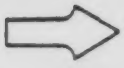
TECHNICAL EDUCATION RESEARCH CENTER — SOUTHWEST
4201 LAKE SHORE DRIVE WACO, TEXAS

COURSE MODULES

COURSE 9 REACTOR AUXILIARY SYSTEMS

MODULE 9-1 Containment and Ventilation of Reactors

MODULE 9-2 Spent Nuclear Fuel Storage and Handling Procedures



MODULE 9-3 Emergency Power Systems for Reactors

MODULE 9-4 Residual and Emergency Cooling Systems for Reactors

Copyright in 1977 by the Technical Education Research Center, Inc. Printed in the United States of America. All rights reserved. These materials, or parts thereof, may not be reproduced in any form without permission.

The materials contained herein were developed pursuant to a contract with the Office of Education, U.S. Department of Health, Education and Welfare: Contract No. 300760356. Contractors undertaking such projects under Government sponsorship are encouraged to express freely their professional judgment in the conduct of the project. Points of view or opinions stated do not, therefore, necessarily represent official Office of Education position or policy.

Technical Education Research Center
Southwest
4201 Lake Shore Drive, Suite 111
Waco, Texas 76710

INTRODUCTION

Any power plant is its own best customer. Each plant uses from 5 to 10 percent of its gross production to support its operation. This power use is distributed between boiler (reactor, in a nuclear plant) and secondary plant items.

A major portion of power goes to operate the reactor circulating pumps (reactor coolant pumps for PWR's and reactor recirculation pumps for BWR's).

In the secondary plant, a major demand comes from main feedwater pumps, condensate pumps and circulating water pumps. For a nuclear plant with a 1300-MWe gross generation, power requirements can approach 130-MWe to support normal plant operation.

In addition to power for operating large equipment, an operator must have power available to operate electrical breakers, reactor protection systems, safety features, actuating systems and instrumentation systems.

To provide for plant operation, several redundant power sources distribute power through separated power buses. These separate power sources and supplies ensure that the reactor will be protected and monitored, and that power will be available to operate emergency equipment.

MODULE PREREQUISITES

The student should have completed courses in 1) nuclear systems, 2) power plant systems and 3) electronics.

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Define:
   - Transmission and distribution network.
   - Switchyard.
   - Substation.
   - Switch gear.
   - Breaker.
   - Disconnect.
   - Bus.
   - Transformer.
2. Identify basic circuits by diagramming an electrical distribution system for a typical power plant.
3. Label a diagram to show normal and alternate power sources in a power plant.
4. Diagram a motor control center and describe its function.
5. Define sources of emergency power.
6. Describe need for redundancy.

DEFINITION OF TERMS

To understand material presented in this section, the student should be able to define the following terms:

Transmission and distribution network - The lines (and all intermediary equipment) used to transport electricity from generator to user.

Substation - That portion of the transmission and distribution network where electricity undergoes change in voltage, and is further separated for distribution to customers.

Switchyard - A substation at a power plant. (This substation is used both to send out and to receive power.)

Switch gear - Equipment used for directing the distribution of power within the power plant.

Breaker switch - A circuit-breaking device used in the distribution of power.

Disconnect - Another method of opening a circuit. Disconnects differ from breakers in that breakers are able to interrupt current flow and are usually remotely controlled. Disconnects, used to open circuits not under load, are generally locally operated.

Bus - A short, rugged power circuit. Switch gear normally consists of buses and breakers.

Transformer - A common device used to either step-up (increase) or step-down (decrease) voltage.

Inplant use of electricity may best be understood by referring to an overall schematic of an inplant distri-

bution system (Figure 9-3-1). This schematic, which includes

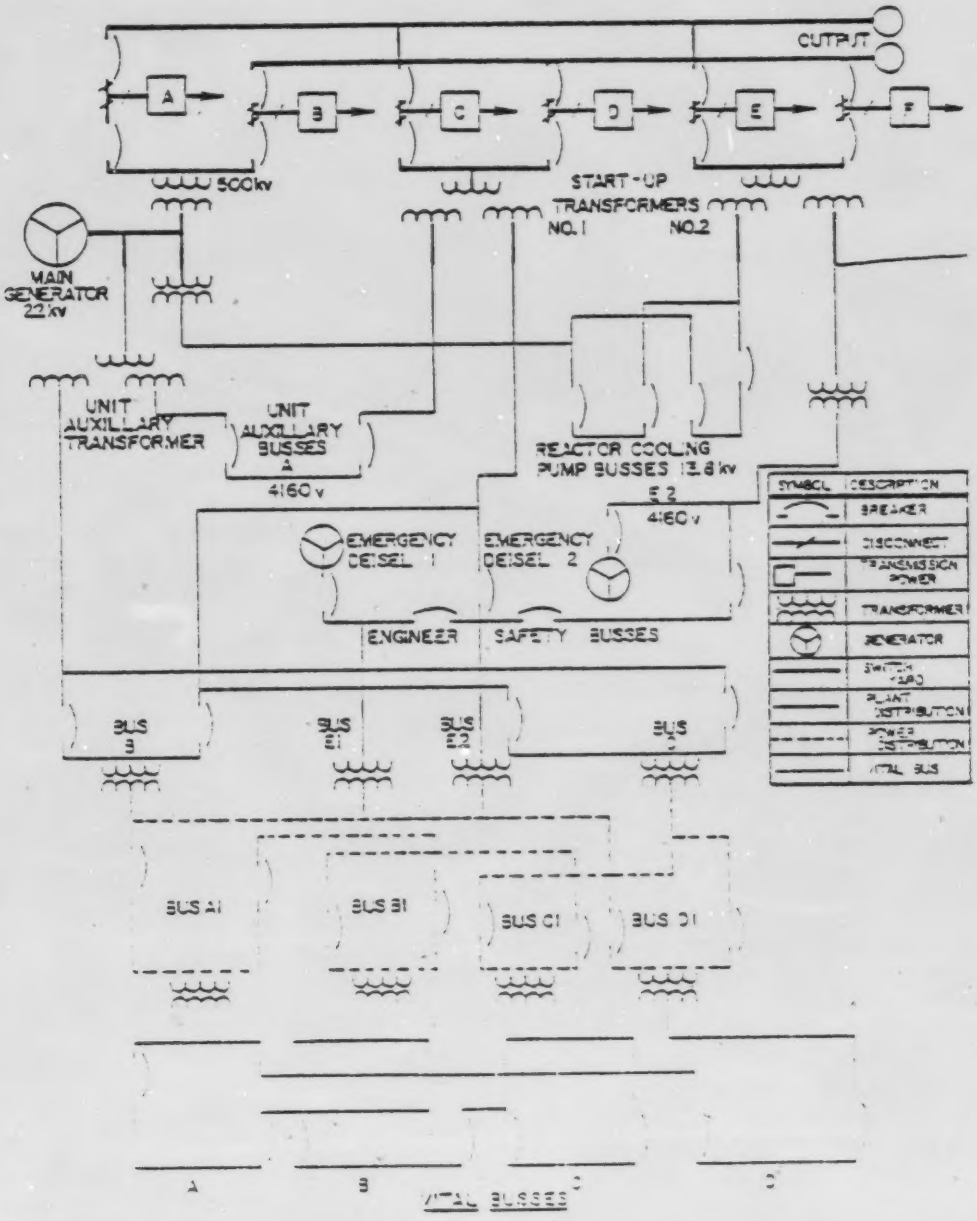


Figure 9-3-1. Internal distribution, reactor power plant.

61

1) the switchyard, 2) the plant distribution system, 3) the motor control system and 4) the vital buses, is drawn so that each subschematic (Figures 9-3-2 through 9-3-5) is a specified portion of it; that is, all the subschematics fit together to form the total inplant distribution system.

PLANT SWITCHYARD

A plant switchyard is the connecting line between a power plant and its transmission and distribution network. Not only is it a system for transmitting power generated by a plant, but also it allows the plant to receive power for operation of auxiliary and emergency equipment when the plant is not generating its own power.

A typical switchyard (Figure 9-3-2) consists of two

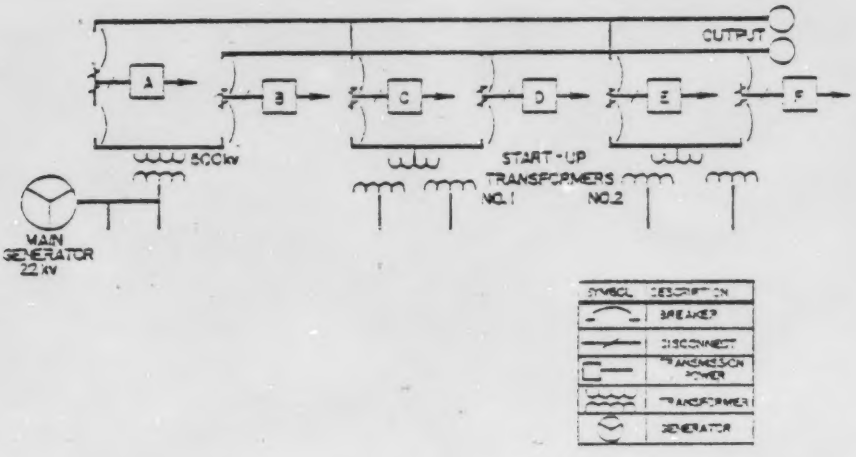


Figure 9-3-2. Typical switchyard.

or more buses connected to three or more transmission lines by the necessary breakers and disconnect switch gear.

PLANT DISTRIBUTION NETWORK

Distribution of power within a power plant (Figure 9-3-3)

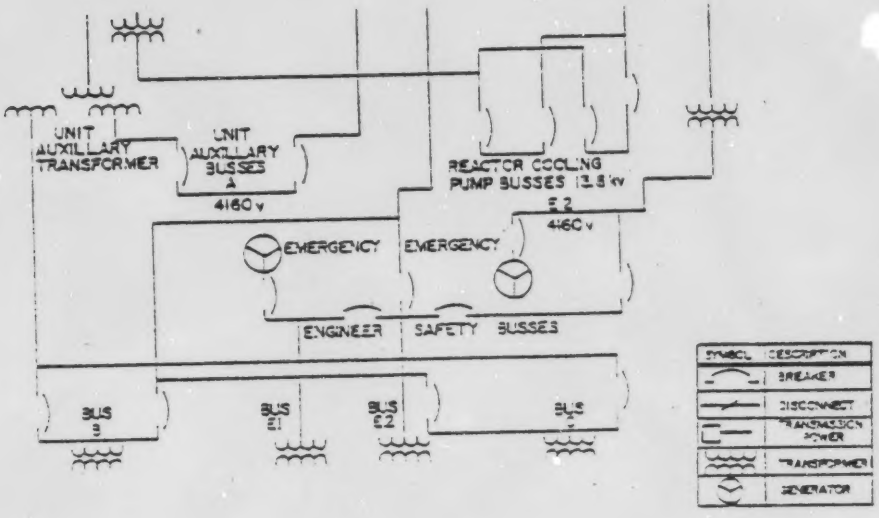


Figure 9-3-3. Plant distribution system.

includes a means of supplying auxiliary power to equipment during normal operation and shut-down operation. Provision must be made to ensure reliability of this power source. Also, a means of de-energizing sections of the network for maintenance while the plant is in operation must be provided.

Each auxiliary power bus has two sources: the switch-yard and the plant generator. Auxiliary power is divided

into three or more buses. The auxiliary equipment is separated such that the plant may continue to operate, at reduced output, with the loss of any one bus. Further, with a planned outage, the plant can operate at full power with one bus de-energized for maintenance.

Safety equipment is powered through emergency buses, designed to receive power from either of two sources: the switchyard or an emergency source - normally a diesel generator, a hydroelectric plant or some other reliable source.

Motor Control Center

Below the plant auxiliary buses, the power distribution to auxiliary equipment is provided by motor control centers. This equipment also is redundant to ensure operating integrity (Figure 9-3-4).

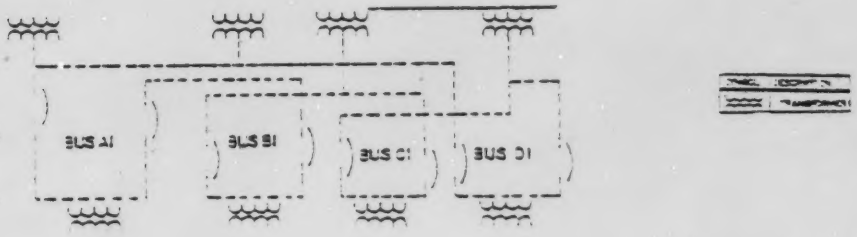


Figure 9-3-4. Motor control center - power distribution.

Vital Power

Beyond the motor control center, power is distributed to protective instrumentation and safety-features control equipment by vital buses (Figure 9-3-5). Each vital bus

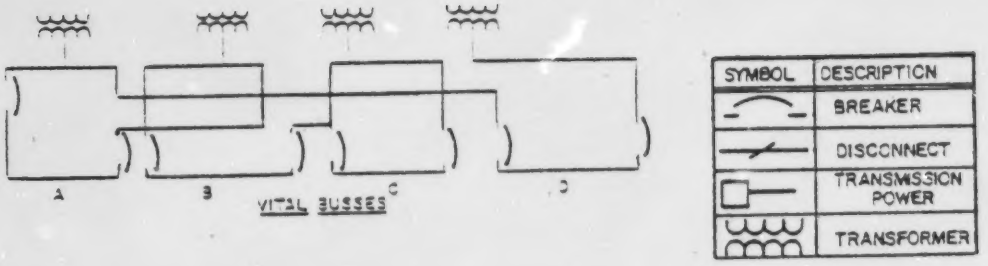


Figure 9-3-5. Vital buses.

has three power supplies: a normal supply; an alternate supply and a d.c. inverter supply.

Each vital bus supplies a reactor protection channel, an instrument bus, and a safety-feature actuation channel. In addition, each vital bus supplies an inverter (a device that supplies d.c. power from an a.c. source, or a.c. power from a d.c. source). Each inverter provides a continuous source of d.c. power to the vital d.c. buses, as well as a floating charge to the battery connected to that bus. In the event of a loss of power to the a.c. vital bus, the inverter provides, from that battery, a.c. power for the vital a.c. power bus.

NORMAL POWER SOURCE

During normal plant operation, the auxiliary buses receive power from the unit auxiliary transformer. The emergency power buses receive power from the plant switchyard. Because equipment served by the emergency buses is safety-feature equipment, and most of it is required only in the event of a plant failure, the unnecessary transfer step

from the unit auxiliary transformer is eliminated.

ALTERNATE POWER SOURCE

During shut-down operation, and until the point in a startup at which the plant generator is producing reliable power, auxiliary power usually comes from another power plant by way of the plant switchyard. This power is stepped down by starting transformers. Then, when power from internal auxiliary sources becomes available (through the auxiliary transformers) the starting transformers are used only as alternate sources. As alternate sources, they are automatically selected if power from auxiliary transformers is interrupted.

EMERGENCY POWER SOURCE

Emergency buses have an alternate power source: diesel generator sets. (Some plants may use gas-trubine, hydroelectric or other sources of backup emergency power.) Diesel generator sets are used most commonly, however.

Diesel generators, sized to handle one train of emergency core cooling equipment and necessary control and instrumentation, provide the operator with a means of monitoring and controlling a plant in a safe, shut-down condition. This power source is used only in the event of a loss of off-site power, however.

Normally, two diesel generators are available. But only one is required to provide the necessary power. The equipment is connected in such a manner as to ensure full redundancy.

A diesel generator starts any time there is a loss of power to either emergency bus, or a loss of off-site pow-

er. It automatically connects to the emergency bus it serves, if there is a loss of power to that bus, as quickly as the generator comes up to speed and voltage (about 10 seconds). Loss of power clears the heavy load from the bus, preventing overloading of the diesel generator when it is connected to the bus. Once connected to the bus, time sequenced starting of heavy equipment averts overload of the diesel generator.

Usually, sufficient fuel oil is stored on-site to provide approximately seven days of diesel operation.

EXERCISES

None.

MATERIALS

Emergency power schematics or drawings of either a PWR or a BWR operating nuclear plant.

PROCEDURES

Use drawings and schematics; write description of happenings and safety procedures that take place following a loss of primary power.

DATA TABLE

None.

REFERENCE MATERIALS

Asimov, Isaac. Inside the Atom. New York: Abelard-Schuman Ltd., 1974.

Glasstone, Samuel. Principles of Nuclear Reactor Engineering. Princeton, NJ: Van Nostrand, 1961.

Hogerton, John F. The Atomic Energy Deskbook. New York: Van Nostrand-Reinhold, 1963.

Nuclear Power and the Environment: Questions and Answers. New York: Unipub, 1973.

Schultz, M.A. Control of Nuclear Reactors and Power Plants. 2nd ed. New York: McGraw-Hill, 1962.

1. List, in descending order from switchyard to the d.c. vital buses, all electrical buses in a power plant:

2. Name power sources for plant auxiliary power buses:

3. Name power sources for plant emergency power buses:

4. Identify normal and alternate power sources for plant auxiliary power buses:

5. Identify normal and alternate power sources for emergency buses:

6. What occurrence causes automatic startup of a diesel generator?

7. What occurrence causes a diesel generator to be connected to the emergency buses?

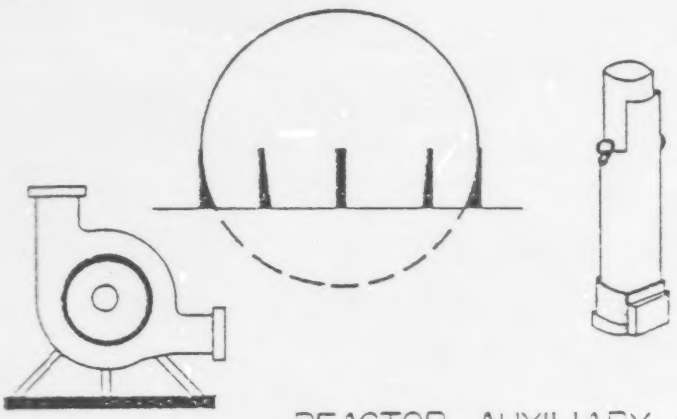
8. How long can a diesel generator normally operate from its on-site fuel supply?

9. Will there be a problem if only one diesel is able to run? If so, what?

10. Sketch a complete plant power system, from switchyard to d.c. vital buses (show reactor coolant pump buses):

NUCLEAR TECHNOLOGY

MODULE 9-4 RESIDUAL AND EMERGENCY COOLING SYSTEMS FOR REACTORS



REACTOR AUXILIARY SYSTEMS

COURSE 9

TEC-SW

TECHNICAL EDUCATION RESEARCH CENTER — SOUTHWEST

4201 LAKE SHORE DRIVE WACO, TEXAS

COURSE MODULES

COURSE 9 REACTOR AUXILIARY SYSTEMS

MODULE 9-1 Containment and Ventilation of Reactors

MODULE 9-2 Spent Nuclear Fuel Storage and Handling Procedures

MODULE 9-3 Emergency Power Systems for Reactors

MODULE 9-4 Residual and Emergency Cooling Systems for Reactors

Copyright in 1977 by the Technical Education Research Center, Inc. Printed in the United States of America. All rights reserved. These materials, or parts thereof, may not be reproduced in any form without permission.

The materials contained herein were developed pursuant to a contract with the Office of Education, U.S. Department of Health, Education and Welfare: Contract No. 300760356. Contractors undertaking such projects under Government sponsorship are encouraged to express freely their professional judgment in the conduct of the project. Points of view or opinions stated do not, therefore, necessarily represent official Office of Education position or policy.

Technical Education Research Center
Southwest
4201 Lake Shore Drive, Suite 111
Waco, Texas 76710

INTRODUCTION

A nuclear reactor is a device used to produce heat, in usable form, by splitting atoms (fission). Production of heat by fission is a natural phenomenon; however, the ability to control this phenomenon and to produce heat in large quantities, at usable levels, has required rapid advances in technology. It is the production of heat - the object of the reaction - that produces some of the greatest difficulties.

To remove the large quantities of heat produced requires large water flow rates at all times - heat produced by the fuel is transferred into the water. And the ability of the reactor coolant system to remove the heat from the reactor is the limiting condition. Also, it is the basis for much of the reactor protection system design. During normal operation, heat removal from fuel equals rate of heat production.

MODULE PREREQUISITES

The student should have completed all previous modules in this Course.

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Sketch and label reactor auxiliary systems.
2. Trace normal flow through these systems.
3. Describe when and why these systems operate.
4. Write brief system descriptions for typical safety injection, emergency cooling and chemical and volume control systems, including a labeled flow diagram.

PRESSURIZED WATER REACTORS

The function of reactor coolant is to cool the fuel and provide a vehicle for transfer of heat to the steam generator. The normal heat-flow path is from fuel pellets through cladding, into coolant, where heat is stored during transport to steam generator.

NORMAL HEAT REMOVAL

In a steam generator, heat stored in reactor coolant is given up to the feedwater. Secondary side fluids continue to extract heat from reactor coolant until temperatures of primary and secondary water are equal, or until primary water exits the steam generator. (Primary water exiting the steam generator is returned to the reactor, where it again picks up heat from the fuel.)

In the steam generator, feedwater is heated to saturation temperature, then additional heat is added (without temperature increase) until all of it is converted into steam.

Heat is stored in the steam until it is given up to the turbine to produce mechanical energy. The turbine cools the steam (and produces, in combination with generator, electrical energy, which is the end product of a nuclear plant).

Once steam has given up as much heat as is practicable, it enters the secondary cycle. There it again becomes water (in the condenser). This water is returned to the steam generator as feedwater.

REACTOR PLANT OPERATION

During reactor operation, close coordination is required between heat production in fuel, transfer of heat in reactor coolant, extraction of heat by steam generator, and use of heat in turbine. Any change in this cycle causes an imbalance in the system.

REACTOR PLANT CASUALTIES

If production of heat in a reactor is greatly reduced (or stops) and heat transfer continues, the reactor suffers a "cooldown," which could result in damage to equipment, components and systems (a heat-extraction accident).

Any reduction in heat removal without an accompanying reduction in heat production can cause a storage of energy in the fuel (or reactor coolant system) that could result in damage to fuel cladding, reactor equipment, components and systems. Loss of heat transfer produces similar results. An LOCA (loss-of-coolant accident) may be caused by 1) a leak in the reactor coolant system, 2) a loss of steam generator feedwater or 3) a loss of turbine load.

REACTOR COOLDOWN

To place a reactor in a condition for maintenance, it is necessary to remove stored heat and heat produced by decay-of-fission products (decay heat). During extended shutdown, it is necessary to provide for continual removal of decay heat.

Initial removal of heat is accomplished by a method called Phase I cooldown. This cooldown is accomplished by

transferring heat from reactor to steam generator, and then into plant circulating water system (Figure 9-4-1).

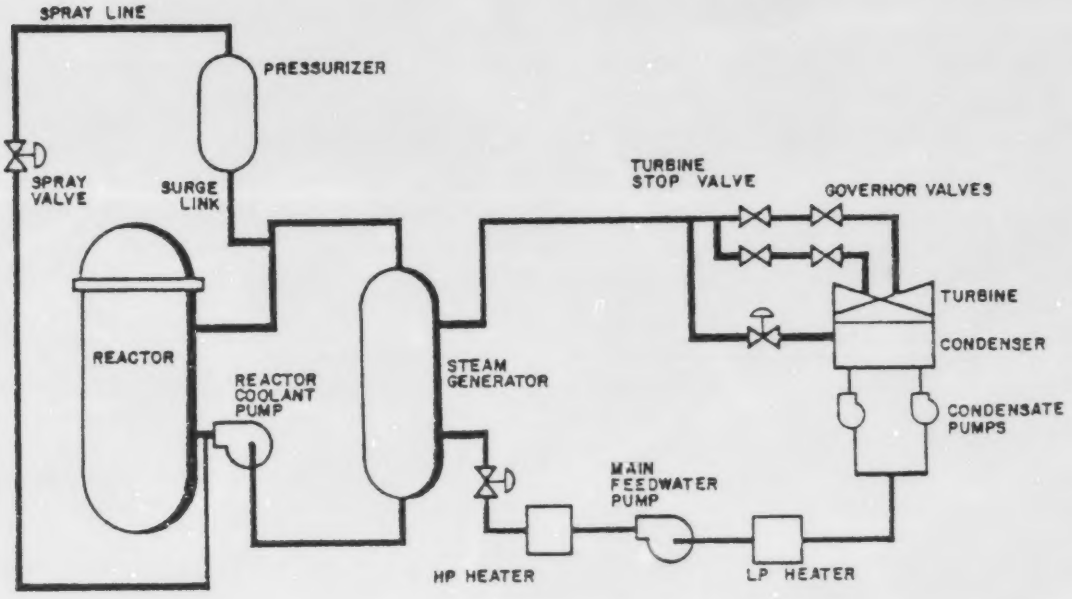


Figure 9-4-1. Flow path for Phase I cooldown.

Early in a Phase I cooldown, driving force, sufficient for heat transfer, exists in the form of high reactor temperature, high steam pressure and low-temperature feedwater.

As cooldown progresses and reactor coolant temperature drops, steam pressure also lowers, and temperature differential between feedwater and reactor coolant decreases. Also, quantity of heat is reduced in each pound of steam. This reduction requires an increase in steam flow to continue cooldown. (With reduced steam pressure, ability to increase steam flow also is reduced.)

When a point is reached at which heat input from reactor coolant pumps and decay heat from core maintain a steady

reactor coolant temperature, Phase I cooldown is complete.

Phase II cooldown consists of 1) establishing a cooling flow path from reactor through the decay heat exchanger and 2) returning cooled water to reactor (Figure 9-4-2).

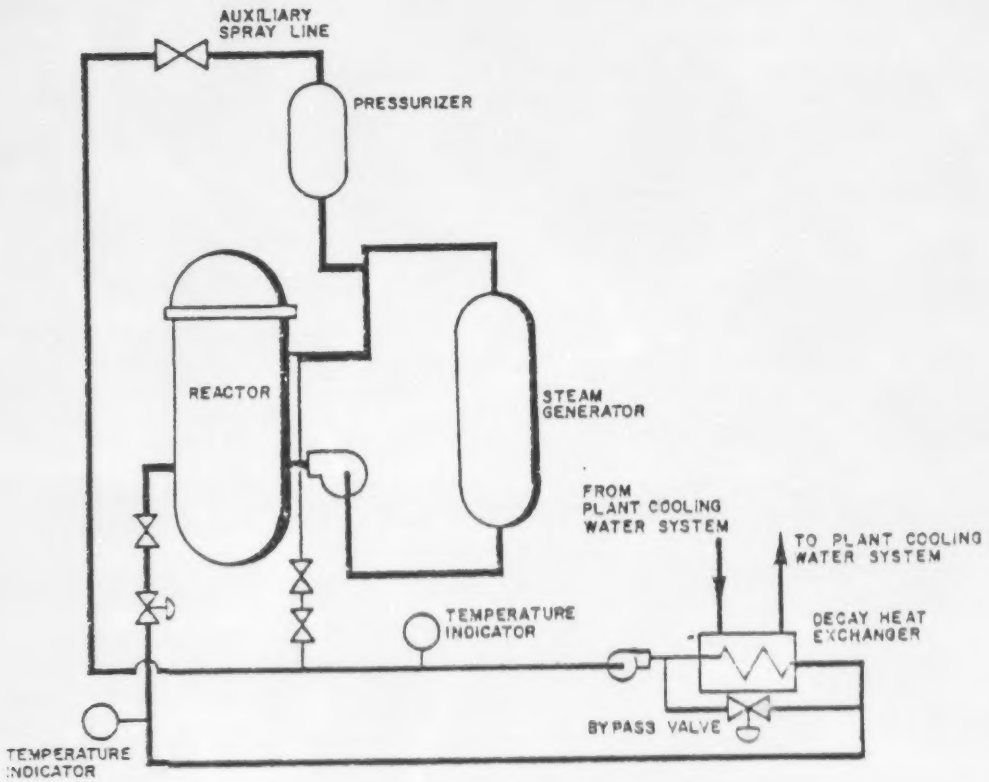


Figure 9-4-2. Flow path for Phase II cooldown.

When reactor coolant temperature and pressure decrease to values compatible with the residual heat removal system.

this system may be placed in operation. Care must be taken to prevent any large reduction in coolant temperature where coolant contacts reactor vessel wall. Flow through the residual heat removal system is started with the decay heat exchanger bypassed. Flow continues in parallel with flow in reactor coolant system until temperature in residual heat removal system is equal to that in reactor coolant system. Only then can the reactor coolant pump be shut down.

Flow in the residual heat removal system is directed through the decay heat exchanger by slowly closing the bypass valve. At this time, operator must 1) observe temperature of water returned to reactor and 2) monitor rate of decrease to ensure that it is within limits prescribed in operating procedures and Technical Specifications.

To understand the concern over cooldown rates, the flow path (Figure 9-4-3) within the reactor vessel must be understood.

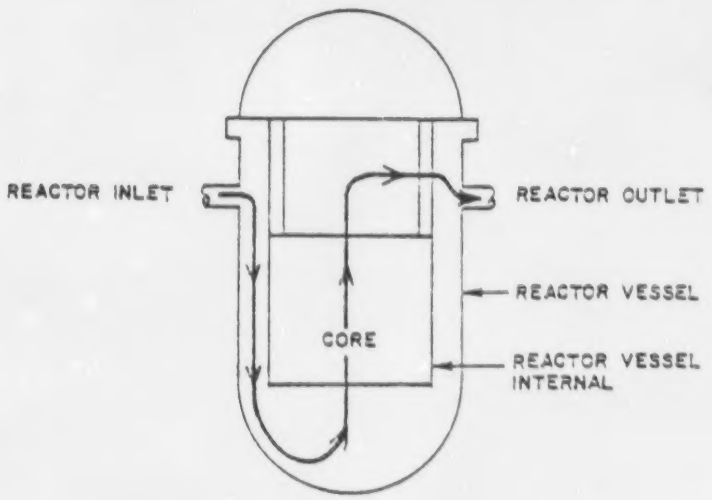


Figure 9-4-3. Flow of coolant within reactor vessel.

Cool water returned to the reactor is directed downward within the reactor by reactor vessel internals; cold water blankets the reactor vessel interior wall, causing interior surface of the vessel wall to cool (Figure 9-4-4). Vessel wall thickness causes delay in cooling of outer vessel wall.

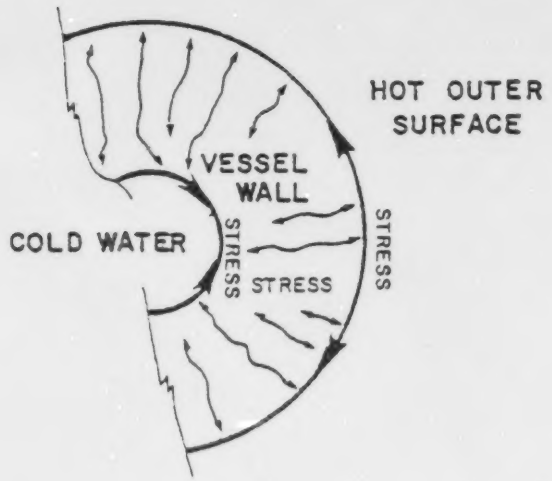


Figure 9-4-4.
Stress generation
in reactor vessel wall.

Rate of cooldown must be controlled to limit total stress on vessel wall (within limits established by the regulatory guidelines). Total stresses are the sum of stresses generated in vessel wall by cooldown and force on the vessel wall caused by internal pressure in the reactor coolant system.

Further limits are placed on rate of cooldown by specific temperature of the reactor coolant system (Figure 9-4-5). An upper pressure below 280°F is based on total stress (below a conservative limit) on vessel brittle fracture tempera-

ture. Above 280°F, a limit increase is caused by a shift in theoretical brittle fracture temperature. The upper limit then becomes a basis of temperature/pressure/stress limit. Cooldown is calculated in stress limit.

During cooldown, to reduce reactor coolant system pressure, the pressurizer must be cooled. The pressurizer is a large tank, constructed of more than 50 tons of metal, containing some 35,000 pounds of water at elevated temperature.

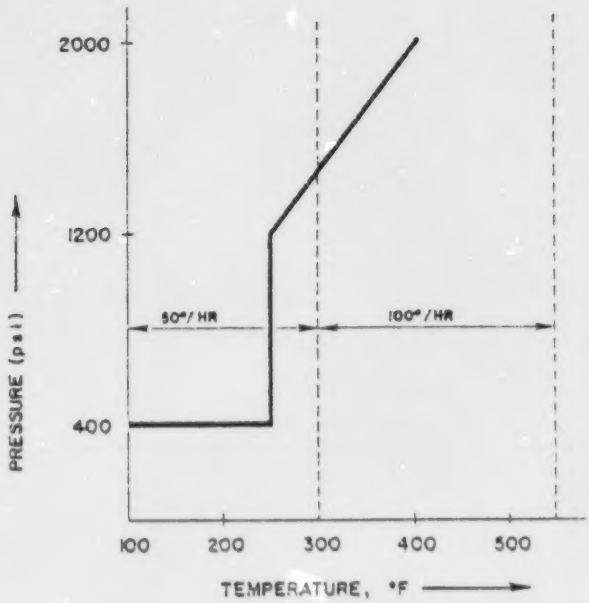


Figure 9-4-5. Reactor coolant temperature versus pressure and cooldown rate.

Cooling is accomplished by circulating reactor coolant through the pressurizer via the pressurizer spray line. This cooling can be accomplished as long as a reactor coolant pump is running to provide a head for flow. Hot water leaves the pressurizer via a surge line, then mixes with the reactor

coolant, giving up its heat to the cooling process. After the reactor coolant pumps are stopped, the pressurizer is cooled by flow from the residual heat removal system (via the auxiliary spray line). At this time, the head for flow is provided by residual heat removal pumps.

Cooldown and depressurization are continued in this manner until reactor temperature is reduced to a desired value, and until the pressurizer is cooled and pressure is reduced to atmospheric. A cooling rate is established at that time for maintaining the reactor at a desired value. Heat (equal to that generated by decay of fission products) is rejected until startup.

REACTOR STARTUP PREPARATION

In the first stage of reactor heatup, cooling water flow through residual heat removal heat exchangers is established, while flow continues through the reactor vessel. Heat produced by decay of fission products, and heat from operation of residual heat removal pumps, increases reactor vessel temperature until operation of reactor coolant pumps is permissible.

EMERGENCY REACTOR COOLING

During an LOCA, heat produced in the fuel has no flow path. Increasing temperature can cause fuel cladding failure and release of radioactive products from fuel.

An LOCA can take may forms, including these more credible ones:

Loss of load on turbine.
Loss of feedwater to steam generator.
Loss of all electrical power (loss of power for reactor cooling pumps).

The reactor must have emergency systems that can provide a flow path for heat from the fuel to a heat sink. (Note: A heat sink is a system or device that can receive and store, or pass on, large quantities of heat - e.g., lakes, cooling towers, suppression pools, spray ponds and the atmosphere.) During normal operation, the heat sink for a generating plant is the main condenser. This condenser receives waste heat from turbine exhaust and passes it to the cooling tower (or lake) through a main circulating water system.

During a partial LOCA, the main condenser provides a heat sink for excess heat; for example, during a partial loss of turbine load, steam is automatically directed to the condenser through turbine bypass valves (Figure 9-4-6).

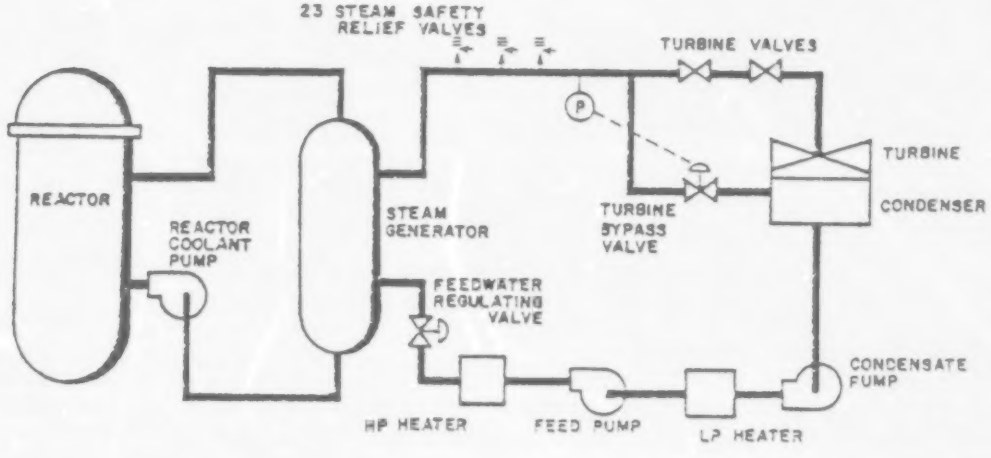


Figure 9-4-6. Flow path for steam during partial LOCA.

Steam header pressure is used to detect a need for additional reactor cooling. An increase in steam header pressure (above a predetermined amount) causes the turbine bypass valve to open and pass steam directly to the condenser. This bypass continues until steam pressure returns to normal.

For a loss of turbine load or turbine trip, the bypass valves lack capacity to pass total steam flow: assistance is required from a steam safety relief valve. When pressure reaches a set point, steam is released to atmosphere until pressure is reduced below a reseating value.

During both conditions (partial or full loss of turbine load) the control system reduces reactor power in a controlled manner to a level compatible with available heat removal.

For more severe LOCA's, there are additional requirements. During a loss of steam generator feedwater accident, a source must be provided for cooling the reactor to a safe level. An emergency feedwater system (Figure 9-4-7) provides this source.

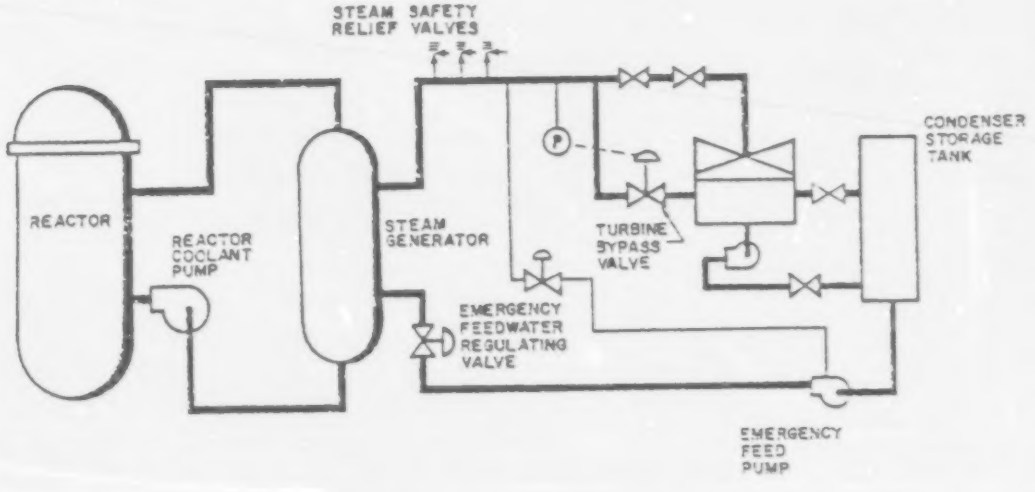


Figure 9-4-7. Emergency feedwater system.

Upon detection of a loss of steam generator feedwater, usually by low feedwater pressure, the emergency feedwater pump starts up, providing cooling for the reactor via the steam generator. Steam is then passed to the condenser via the turbine bypass valve. During this accident, reactor protection system trips reactor to protect fuel and reactor coolant system. Shutdown is required because the emergency cooling system is sized to provide only for heat generated by fission decay, not heat produced by reactor power.

During a loss of all electrical power (including power for reactor coolant pumps), the emergency feedwater system provides for cooling of reactor via the steam generator. Cooling the steam generator causes natural circulation within the reactor coolant system (without assistance from pumps). Also, the emergency feedwater pump is steam driven and requires no support system for operation. Steam generated is then released to the atmosphere via steam safety relief valves. There is sufficient water available in the condensate storage system to provide for eight hours of emergency feedwater system operation.

LOSS OF REACTOR COOLANT ACCIDENT (LOCA)

An LOCA can take many forms, from small packing valve leaks to rupture of a main reactor coolant pipe. Because a method must be provided to ensure cooling of fuel during any size LOCA, several systems are provided:

Normal reactor coolant makeup system - for small leaks.

High-pressure injection system - for small to intermediate leaks.

Low-pressure injection system - for intermediate to large leaks.

Core flooding systems - for large leaks.

During normal reactor operation, a small amount of leakage can be accepted (within limits set by Technical Specifications) if it does not present a hazard to the general public. Such leakage must be closely monitored and accounted for by normal operating procedures. If it becomes excessive, the reactor must be shut down, cooled down and depressurized, and the leakage repaired. Normal volume control is designed to provide for this leakage.

For intermediate leakage (a rate greater than capacity of volume control and makeup system), a high-pressure injection system maintains reactor volume and pressure until cooling and depressurization of reactor can be accomplished by normal means.

For intermediate to large leakage, a low-pressure injection system maintains reactor volume and pressure until cooldown can be accomplished.

Large leakage may cause rapid depressurization of the reactor coolant system and large loss of volume before a mechanical system can provide makeup water. Core flooding refills the reactor vessel until high-pressure and low-pressure injection systems can establish continuous flow through the core and out the pipe rupture (Figure 9-4-8).

During an LOCA, the reactor is tripped when loss of water is great enough to reduce reactor coolant pressure to a set trip point. High-pressure injection system starts when its set point is reached; this start includes starting of pumps, opening of suction valves from refueling storage tank, and opening of injection valves into reactor coolant system.

A refueling water storage tank supplies the large volume of water required to make up for water losses from the reac-

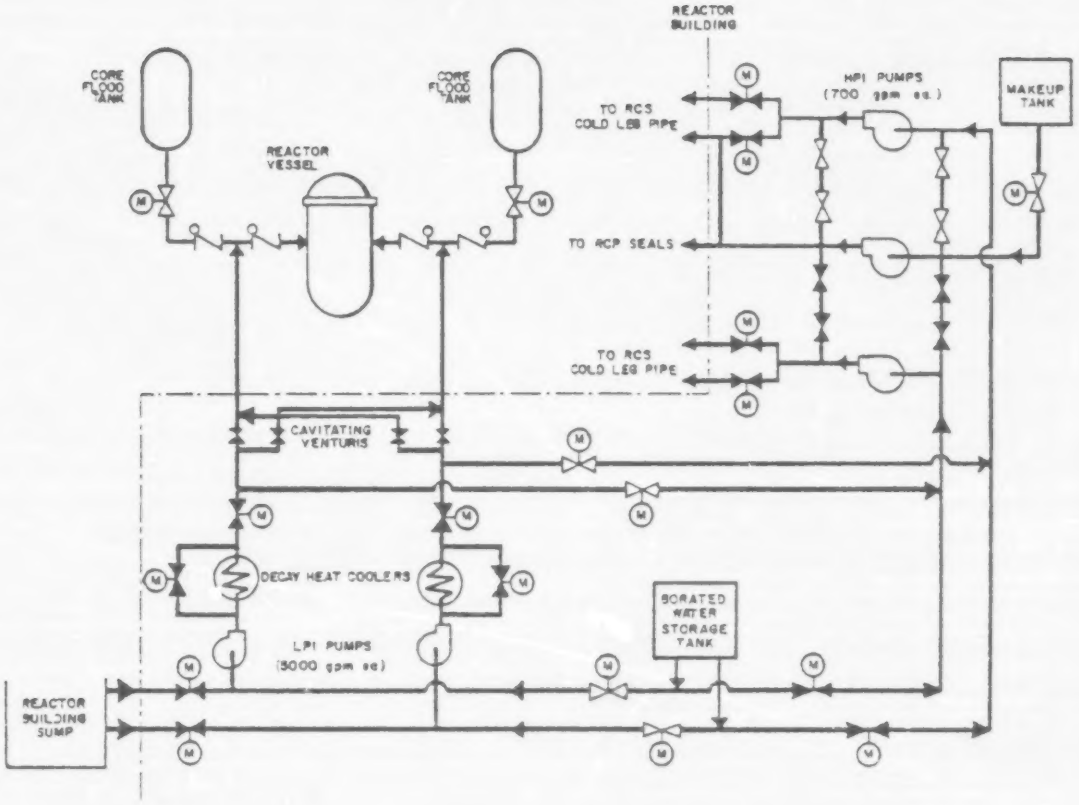


Figure 9-4-8. Emergency core cooling system.

tor coolant system. High-pressure injection pumps provide the necessary head to overcome reactor coolant system pressure losses. High-pressure injection valves provide necessary flow paths for 1) a large volume flow rate and 2) in-

sufficient number to ensure that failure of any one flow path or valve or pump does not render system inoperative. If starting the high-pressure injection system stabilizes reactor coolant volume and pressure, the operator then may control system to establish normal cooldown.

If operation of high-pressure injection system does not stabilize volume and pressure, decreasing pressure starts low-pressure injection system. Starting the low-pressure injection system includes starting of pumps, opening of suction valves (from refueling water storage tank), and opening of discharge valves to reactor.

If cooling of reactor takes more water than is available from storage tank, the low-pressure system can take water from the reactor containment building sump and return it to the reactor or to the high-pressure injection pump suction (through the residual heat-removal exchangers). This capability provides long-term cooling of the reactor after an LOCA.

BOILING WATER REACTORS

Normal heat removal from a BWR is accomplished much the same; however, a BWR uses a combined primary and secondary cycle; water used to cool fuel also becomes the steam that drives the turbine.

The fact that the same steam used in the turbine is also returned to the reactor as feedwater leads to impurities being introduced into the reactor. Because primary water boils in the reactor vessel, a concentration of impuri-

ties takes place there. To limit this chemical concentration, reactor water cleanup is provided (Figure 9-4-9).

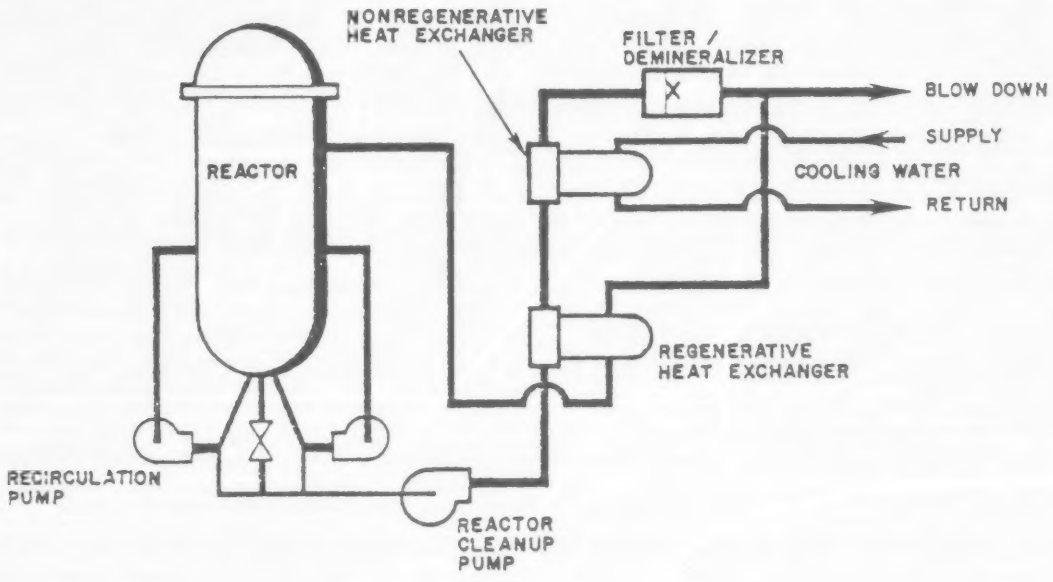


Figure 9-4-9. Reactor water cleanup system.

Bypass flow, taken from recirculation pump suction, is pumped through regenerative and nonregenerative heat exchangers, then through ion exchangers. From the ion exchangers, the flow can be directed to either the reactor feedwater system, the condenser or the waste disposal system.

During reactor coolant water expansion, with low steam flow, reactor coolant volume is controlled by processing any excess through the reactor water cleanup system to the condenser or waste disposal system.

EMERGENCY SHUTDOWN

If control rods are unable to shut down the reactor, a chemical (neutron poison) is introduced into the reactor coolant system. A standby liquid control system (Figure 9-4-10) provides a flow path of soluble poison from storage tank to reactor.

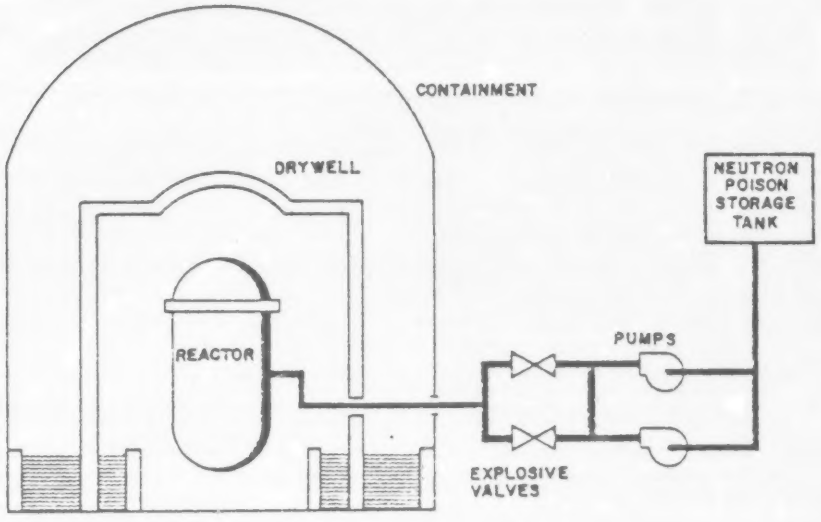


Figure 9-4-10. Standby liquid control system.

Two full-capacity pumps provide the required flow rate (to change from hot operation to cold shutdown), and to keep the reactor shut down under all credible conditions.

The storage tank contains sodium pentaborate solution. The boron in this compound absorbs neutrons, providing for shutdown of nuclear reaction. The two full-capacity, positive-displacement pumps provide head to pump the solution into the reactor vessel. Two full-flow, explosive valves provide redundance to overcome single failure.

LOSS OF LOAD

During BWR operation, an LOCA can lead to serious fuel damage and to release of fission product activity. To prevent this release, and to limit fuel damage, a reactor core isolation cooling system is provided (Figure 9-4-11).

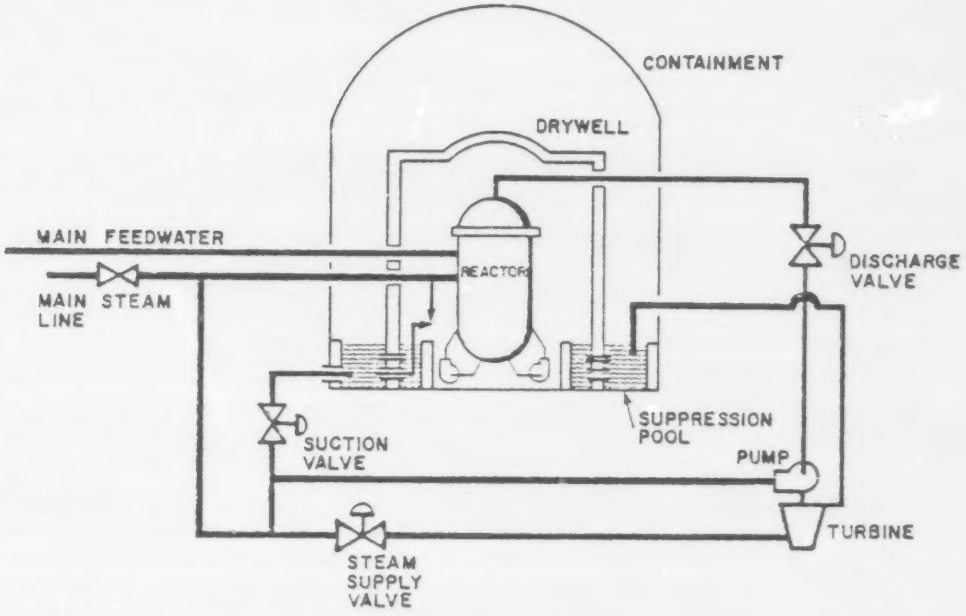


Figure 9-4-11. Reactor core isolation cooling system.

During a loss-of-load accident, when condenser is unavailable for heat removal, reactor core isolation cooling system removes heat from fuel and cools reactor. Water stored in the suppression pool is pumped (by a turbine-driven pump) into the reactor where it is turned into steam to drive the pump turbine. Heat not used to drive the pump

is discharged into the suppression pool via safety relief valves. This self-sustaining cycle can maintain reactor fuel temperature within an acceptable range until the condenser can be returned to service. Also, this system provides safe shutdown and cooldown of reactor in the event of a loss of normal feedwater or electrical power. Water from the suppression pool is used as a source of feedwater and as a heat sink for decay heat in the reactor.

EMERGENCY CORE COOLING SYSTEMS

The emergency core cooling systems provide at least two independent cooling systems. Each can function (without offsite power) to prevent fuel cladding failure for any failure, including a break (equivalent in size to the largest pipe used in the plant).

These design objectives are met by combined automatic depressurization high-pressure core spraying - low-pressure core spraying and low-pressure coolant injection systems (Figure 9-4-12).

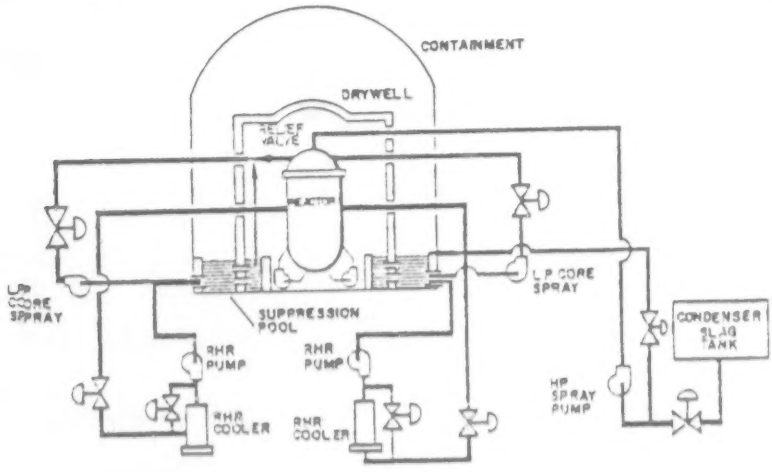


Figure 9-4-12. Emergency core cooling system.

HIGH-PRESSURE CORE SPRAY

A high-pressure core spraying system takes water from the condensate storage tank (or suppression pool) and sprays it into the reactor vessel above the fuel. By providing high-pressure spray water above the assemblies, this system can provide cooling for fuel assemblies during a number of different accidents, regardless of size of break or reactor water level.

LOW-PRESSURE CORE SPRAY

A low-pressure core spraying system provides core fuel cooling once reactor pressure is reduced. By delivering water above the fuel assemblies, it can provide fuel cooling for a large variety of accidents, regardless of water level in reactor. The water source for core cooling (low-pressure core spraying) is the suppression pool.

AUTOMATIC DEPRESSURIZATION SYSTEM

An automatic depressurization system relieves pressure in the reactor by operating selected safety relief valves. This system removes heat while relieving steam to the suppression pool as one or more of the water systems provides feedwater to the reactor vessel.

LOW-PRESSURE COOLANT INJECTION

The residual heat removal system, when operated in the low-pressure coolant injection mode, delivers suppression pool water to the reactor vessel inside the shroud area.

This water can be cooled by the residual heat removal cooler to provide heat-sink residual heat removal cooling water.

Note: Depending upon size of accident, and availability of components and power supplies, operation of a combination of these systems provides necessary heat removal and feedwater to protect fuel cladding against overheating and failure.

EXERCISES

1. Sketch and describe a normal reactor heat-flow path.
2. List function(s) of the following systems:

   Primary coolant.
   Main steam.
   Main feedwater.
   Emergency feedwater.

3. Identify source of makeup water for an LOCA.
4. Identify system(s) providing protection against the following:

   Small leaks.
   Small to intermediate leaks.
   Intermediate to large leaks.
   Large leaks.

5. What is purpose of emergency cooling systems?
6. Describe residual heat removal system functions (normal and emergency).

7. Define "heat sink;" list other heat sinks used in a reactor plant.
8. What events take place when high-pressure injection is actuated?
9. List systems used in a BWR to provide emergency core cooling; where possible, identify system that provides a similar function in a PWR.
10. How is decay heat removed from a BWR during loss of all electrical power?

MATERIALS

None.

PROCEDURES

Write a complete, but brief, description for 1) a typical safety injection system, 2) an emergency cooling system and 3) a volume and chemical control system. (Include flow diagrams.)

DATA TABLE

None.

REFERENCE MATERIALS

Babcock and Wilcox Co. Babcock - 205NSS Design Summary. Lynchburg, VA: B & W, 1976.

_____. Steam. 38th ed. New York: B & W, 1976.

Combustion Engineering, Inc. System 80. Windsor, CT: Combustion Engineering, Inc., 1976.

General Electric Co. (GE). BWR-6, General Description of a Boiling Water Reactor, NEDP-20120. 14th ed. San Jose, CA: GE Nuclear Energy Division, April 1975.

Sneen, J.C. Nuclear Power Plant Systems and Equipment. New York: Industrial Press, 1972.

1. Briefly describe the following systems (use appropriate line drawings):

   Reactor cooldown, Phase I.

   Reactor cooldown, Phase II.

   Loss of turbine load.

   Loss of steam generator feedwater.

   Emergency core cooling system.

   Reactor water cleanup system.

   Standby liquid control system.

Reactor core isolation cooling system.

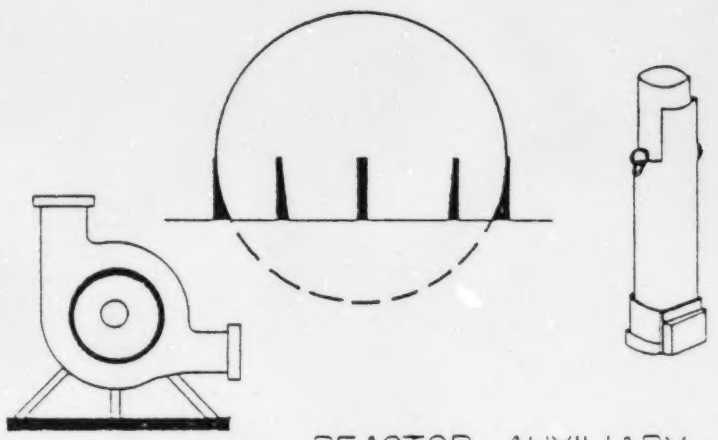
2. Use a system diagram provided by the instructor; trace flow paths during normal operations; add flow arrows; label all components and valves.

3. From answers given for question 1, describe why and when each of these systems come into operation.

NUCLEAR TECHNOLOGY

INDEX

COURSE 9 - REACTOR AUXILIARY SYSTEMS



TERC-SW

TECHNICAL EDUCATION RESEARCH CENTER — SOUTHWEST
4201 LAKE SHORE DRIVE WACO, TEXAS

NUCLEAR TECHNOLOGY SERIES

INDEX

for

COURSE 9, REACTOR AUXILIARY SYSTEMS (FOUR MODULES)

NOTE: Each module in this Course is numbered separately, and the page numbers begin with page 1 in each module. Therefore, this Index lists for each entry the module number (underlined), followed by a comma and the page number. Figures and tables are listed by their numbers, followed by the module and page number in parentheses. A Master Index is available in a separate volume. This general index merges all index entries from the 35 Nuclear Courses. It is a valuable research tool, with which the student or researcher can locate all the references to any subject within the 35 Courses.

Auxiliary power
  motor control center for, 9-3, 7

Breaker switch
  definition of, 9-3, 3
Bridge
  fuel-storage handling, 9-2, 11
  fuel-handling, 9-2, 5-10
Buses
  definition of, 9-3, 3
  vital, power distribution by, 9-3, 8

Containment
  access to, 9-1, 14, 15
  for boiling water reactor, General Electric Mark I design of, 9-1, 7, 8; General Electric Mark II design of, 9-1, 9, 10; General Electric Mark III design of, 9-1, 10, 11
  construction, 9-1, 14
  design of, 9-1, 11, 12
  penetrations of, 9-1, 16; Fig. 9-1-6 (9-1, 17)
  for pressurized water reactor, 9-1, 5, 6
  reactor, final barrier, 9-1, 1; safety zones of, 9-1, 1
Coolant flow
  during reactor startup, 9-4, 10
Cooldown
  generation of stress by, 9-4, 8
  phase I of, 9-4, 4-6
  phase II of, 9-4, 6
  rate of, 9-4, 8, 9
  and residual heat removal system, 9-4,

Design basis accident
  definition of, 9-1, 13
Disconnect
  definition of, 9-3, 3
Distribution
  network, reactor plant, 9-3, 6, 7
  system, power plant, 9-3, 1, 3

Emergency Core Coolant System
  purpose of, 9-4, 14-16
  types of, 9-4, 20-22

Fuel
  cladding, 9-1, 1
  handling, elevator, 9-2, 14; system, 9-2, 3, 4
  hoist, 9-2, 8-10
  transfer system, 9-2, 11-13

Heat removal
  in boiling water reactor, 9-4, 16, 17
  in loss of coolant accident, 9-4, 11-14
  normal, 9-4, 3
  and power plant accidents, 9-4, 4
  system, containment, 9-1, 17
Heat sink
  in loss of coolant accident, 9-4, 11
Hydrogen
  post-accident removal of, 9-1, 19

Ice condenser containment
  for pressurized water reactor, 9-1, 6,

Loss of coolant accident
  definition of, 9-1, 11
  forms of, 9-4, 10, 11
  phases of, 9-1, 11, 12
Loss of load accident
  reactor core isolation cooling system in, 9-4, 19

Power source
  alternate, 9-3, 9
  emergency, 9-3, 9, 10
  normal, 9-3, 3
Pressurizer
  cooling of, 9-4, 9, 10

Refueling
  canal, 9-1, 4,5
  procedures, 9-2, 15-17

Shutdown
emergency, 9-4, 18
Substation
definition of, 9-3, 3
Switch gear
definition of, 9-3, 3
Switchyard
definition of, 9-3, 3
reactor plant, 9-3, 5

Transformer
definition of, 9-3, 3
Transmission and distribution network
definition of, 9-3, 3; Fig. 9-3-1 (9-3, 4); Fig. 9-3-3 (9-3, 6)

Ventilation radiation monitor
containment, 9-1, 18
Vessel
reactor, as second containment barrier, 9-1, 1